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NEUTRON DOSIMETERS AND SURVEY METERS IN ACCELERATORS, REACTORS AND OTHER NEUTRON ENVIRONMENTS

by

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NEUTRON DOSIMETERS AND SURVEY METERS IN ACCELERATORS, REACTORS AND OTHER NEUTRON ENVIRONMENTS

A report prepared by SENES Consultants Limited under contract to the Atomic Energy Control Board.

ABSTRACT

Neutron fields in occupationally accessible areas around nuclear reactors, radioisotope sources and medical and high energy accelerators have been characterized using currently available information. Neutron rem meters, such as the Leake detector, are the most suitable instruments available for conducting neutron dose rate surveys in the vicinity of radioisotope neutron sources, nuclear reactors and medical accelerators. However, these instruments have been shown to be deficient in that they overrespond by a factor of four to neutrons in the 0.1 to 1 MeV range and are insensitive to neutrons from about 1 eV up to about 10 keV. Also, they are insensitive to neutrons above 20 MeV and their use must be restricted near high energy accelerators where significant numbers of neutrons above 20 MeV are known to be present. The most suitable instrument ot measure dose from neutrons above 20 MeV is the $^{12}C(n,2n)^{11}C$ scintillation chamber. Commercially available rem meters frequently use BF_3 counters in the pulse mode to detect thermal neutrons. Therefore, measurements around pulsed accelerators must be made with caution to ensure that the detector is not saturated during each pulse and that the accelerator pulse period is greater than the response time of the detector.

The personal neutron dosimeters currently available either are known to be insensitive to neutrons above 20 MeV or have not been tested. Also, all except the albedo dosimeter are insensitive to or have not been tested for neutron energies in the range 1 eV to 10 keV. Several dosimeter types respond reasonably well to neutrons in the energy range 0.1 to 15 MeV, for example, CR-39, bubble and superheated drop detectors. However, the first has a lower limit of sensitivity of about 0.3 mSv. The bubble detector can be designed to measure doses as small as 1 μSv and offers the additional benefit of direct-reading capability. The superheated drop detector is not suitable for use around pulsed accelerators because the real time analysis of bubble formation introduces a dead time relative to the pulsed source that the wearer cannot monitor.

RÉSUMÉ

Les champs de neutrons dans les aires accessibles aux travailleurs autour des réacteurs, des sources de radio-isotopes, des accélérateurs médicaux et à haute énergie ont été caractérisés grâce aux données couramment disponibles. Les radiamètres à neutrons en rem, comme le détecteur Leake, sont les instruments disponibles les plus appropriés pour faire des relevés du débit d'émission de neutrons près des sources de neutrons de radio-isotopes, des réacteurs nucléaires et des accélérateurs médicaux. Cependant, ces appareils se sont révélés inadéquats car ils répondent de façon exagérée par un facteur de 4 aux neutrons dont l'énergie varie entre 0,1 et 1 MeV. De plus, ils sont insensibles aux neutrons de 1 eV à 10 keV au-dessus de 10 MeV. Leur usage doit

aussi être limité aux environs des accélérateurs à haute énergie où l'on sait que de nombreux neutrons d'énergie supérieure à 10 MeV sont présents. L'appareil le plus approprié pour mesurer les doses de neutrons au-dessus de 20 MeV est la chambre à scintillations au $^{12}C(n,2n)^{11}C$. Certains radiamètres en rem disponibles sur le marché utilisent fréquemment des compteurs BF_3 en mode impulsion pour déceler les neutrons thermiques. Par conséquent, les mesures autour des accélérateurs à impulsions doivent être faites avec précaution afin de s'assurer que le détecteur n'est pas saturé durant chaque impulsion et que la période d'impulsion de l'accélérateur est plus gradne que le temps de réponse du détecteur.

Les dosimètres à neutrons individuels présentement disponibles sont insensibles aux neutrons au-dessous de 20 MeV ou n'ont pas encore été éprouvés. De plus, à l'exception du dosimètre albédo, tous les dosimètres sont insensibles aux neutrons dont l'énergie se situe entre 1 eV et 10 keV ou qui n'ont pas encore été éprouvés. Plusieurs types de dosimètres répondent raisonnablement bien aux neutrons dont l'énergie se situe entre 0,1 et 15 MeV, comme le CR-39, les détecteurs à bulles et les détecteurs de chute à surchauffe. Le détecteur à bulles peut être conçu pour mesurer des doses aussi faibles que 1 μSv et offre l'avantage supplémentaire d'une lecture directe. Le détecteur de chute à surchauffe ne convient pas pour des usages autour des accélérateurs à impulsions, car l'analyse en temps réel de la formation de la bulle introduit un temps mort par rapport à la source d'impulsion, que le porteur ne peut contrôler.

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GLOSSARY

ABSORBED DOSE	D, is the quotient of de by dm where de is the mean energy imparted by ionizing radiation to matter of mass dm.
KERMA	K, is the quotient of dE_{tr} by dm, where dE_{tr} is the sum of the initial kinetic energies of all the charged ionizing particles liberated by uncharged ionizing particles in a material of mass dm.
LETHARGY	u, is a dimensionless index representing neutron energy E; $u = \ln (E_o/E)$, where E_o is taken as 10 MeV. An energy flux $Q(E)$ can be converted to a lethargy flux $L(u)$ by $L(u) = E.Q(E)$.
LINEAL ENERGY	y, has been defined as the energy e imparted to the matter in a volume by a single energy-deposition event, divided by the mean chord length l through that volume (of a counter): $y = e/l$.
LINEAR ENERGY TRANSFER	LET, of a charged particle in a medium is the quotient of dE by dl where dl is the distance traversed by the particle and dE is the mean energy loss due to collisions with energy transfers less than some specific value Δ. When Δ is unrestricted, all possible energy transfers are included and the LET_{∞} is synonomous with collision stopping power.

1.0 INTRODUCTION

Many of the facilities licensed by the Atomic Energy Control Board (AECB) produce neutron radiation fields in occupationally accessible areas. The accurate measurement of these fields and reliability of personal neutron monitoring systems with the necessary energy response, sensitivity and accuracy are of concern to the AECB. Consequently, a study to evaluate instruments and equipment for measuring neutrons was contracted by the AECB. The results are reported below.

The objective of this study was to determine which type(s) of neutron dosimeters for personal dose assessments and survey meters for radiation exposure control is (are) "best suited" for use at certain nuclear facilities (mainly particle accelerators) and in some applications involving radioisotopes where measurable neutron doses can be received by workers. "Best suited" refers to best energy response (flatness of energy response curve) over the energy range of neutrons responsible for (most of) the dose received and least affected by the environmental conditions (e.g. temperature, humidity, rf fields, etc.) at the work sites. In addition, ease of calibration, accuracy and reliability of each instrument are assessed in this work.

The scope of work included discussions with AECB project officers and licensee staff at identified facilities and with specific radioisotope users, and a thorough search of the open literature, to obtain spectral information on the neutron fields at occupied areas in the work place. Computer searches of commercial data bases proved inefficient as many inappropriate papers were identified that relate to neutron detection in scattering experiments of fusion facilities etc. Consequently, the following known sources of appropriate literature were searched directly; Health Physics Journal, Radiation Protection Dosimetry, Nuclear Tracks, Physics in Medicine and Biology, and Medical Physics. The characteristics of neutron dosimeters and survey meters were obtained from the manufacturers of the instruments and from the open literature. Information that is not available from the above sources is identified and laboratory and field testing programs that would provide such information are outlined.

The main body of this report is organized into five chapters not including the Introduction.

Chapter 2 is titled "Characteristics of Neutron Fields in Occupationally Accessible Areas". Temporal and spatial variations in neutron field strengths in occupationally accessible areas around accelerators, reactors and radioisotopic neutron sources are described. Neutron energy spectra and intensity are presented where information is available.

Chapter 3 is titled "Portable and Fixed Neutron Monitors". Neutron detection systems applicable for area monitoring but unsuitable for personal monitoring are described. Instruments are characterized with respect to energy response, directionality, response time, and sensitivity to interference from other radiation, temperature and humidity.

Chapter 4 is titled "Personal Neutron Dosimeters". Small, portable neutron detectors requiring no external power source and suitable for personal monitoring are described. The dose-equivalent energy response, shelf life, impact resistance and background are reviewed in addition to the characteristics listed in the paragraph above.

Chapter 5 is titled "Assessment of Best Suited Neutron Monitors and Dosimeters". The objective of this comparison is to evaluate neutron detection equipment with respect to the unique characteristics of the various neutron sources identified in Chapter 2. Instances where information is insufficient or missing altogether are noted.

Chapter 6 is titled "Recommendations." Experimental research and development projects are recommended that would provide information that has been identified as insufficient or lacking in the previous chapter. Neutron detection systems that have the highest potential for providing a complete and accurate system of monitoring are identified.

The final section is titled "References and Bibliography." An extensive list of current research papers, journal articles and reports is compiled in support of this report and as supplementary reference material.

2.0 CHARACTERISTICS OF NEUTRON FIELDS IN OCCUPATIONALLY ACCESSIBLE AREAS

Stray neutron radiation fields are found in occupationally accessible areas in the vicinity of high energy accelerators, nuclear reactors and radioisotopic neutron sources. Based upon measurements at representative Canadian facilities and information gathered from the open literature, neutron dose equivalent rates and energy spectra of these neutron fields are characterized.

2.1 Accelerators

2.1.1 Medical Accelerators

Neutrons are produced in materials commonly found in the construction of medical electron accelerators when x-ray energies exceed the photoneutron threshold at about 6 MeV. There are currently several types of accelerators installed at facilities across Canada which produce photons in excess of this critical value:

- Clinac-20 and Clinac-35 which produce 20 and 35 MeV photons, respectively, manufactured by Varian
- Therac-20 and Therac-25 which produce 20 and 25 MeV photons, respectively, manufactured by Atomic Energy of Canada Limited;
- Philips SL-25 and SL-75-14, which produce 25 and 10 MeV photons, respectively, manufactured by Philips Medical Systems;

to list a few.

Uwamino (1986) measured neutron spectra and dose equivalent outside two shielding doors at an electron cyclotron (medical microtron, MM22 manufactured by Instrument AB Scanditronix, Uppsala, Sweden). The electrons were accelerated to 21 MeV, pulsed with a width of 4 us and a period of 4 ms. Access to the shielded room was through two doors at either end of a maze. The inner door was constructed of 11.5 cm thick polyethylene. The outer door consisted of 10 cm thick polyethylene with a 5 cm thick lead liner on the outside and two 0.5 mm thick cadmium plates inserted in the centre of the polyethylene and at the interface between the polyethylene and lead. The neutron spectra were measured with a high efficiency ^{3}He counter spectrometer and moderating spheres of diameter 23, 15, 11, 8.1 and 0 cm. The neutron energy spectra in the occupationally accessible area outside the shielding doors are shown in Figure 2.1-1. These were the only neutron spectra measured near medical accelerators that were found in the literature. The dose equivalent was measured with the Studsvik 2202D rem counter (Studsvik Energeiteknik AB). Neutron dose equivalent rates ranged from 0.046 uSv/x-ray Gy with both doors closed to 10 uSv/x-ray Gy with both doors open. The beam intensity had been reduced to 0.01 to 0.1 Gy/min to eliminate pileup effects, in measurements by the ^{3}He and the rem counters.

The Ontario Cancer Institute at the Princess Margaret Hospital in Toronto operates four medical accelerators used for cancer therapy. The results of stray neutron dose measurements near one of the units, Therac-25, are reported here (Rawlinson, 1986). These results were measured at 1 m from the head while the accelerator was operating in the 25 MV x-ray mode at a dose rate of 250 cGy/min. Corresponding radiation levels produced while the machine is operating in the electron modes are negligible by comparison. Neutron levels were measured using a Nuclear Research Corporation Model NP-2 (Andersson-Braun "Snoopy" type) rem survey meter calibrated at the National Research Council

(NRC), Ottawa. The minimum detectable reading on the instrument was 0.5 uSv/h. In areas accessible to personnel during accelerator operation, significant readings of typically 2 uSv/h were obtained only in the Therac mezzanine. Also, the shielding door attenuates the neutron dose rate outside the entrance to the maze by about a factor of 200, from 100 to 0.5 uSv/h. In the past, all exposures to personnel as measured by the neutron film badge service provided by the Department of National Health and Welfare (DNHW) were reported to be less than the detection limit, 0.2 mSv. Eventually, personal neutron monitoring was discontinued, (Rawlinson, 1986).

The Victoria Cancer Clinic operates a Philips 75-14 linear accelerator for radiation treatment of patients. In the x-ray mode it delivers photons up to 10 MeV and in the electron mode electron energies can be varied from 4 to 14 MeV. Normal concrete (density 2.35 g/cm^3) provides the neutron shielding between the source and the accessible areas. Neutron dose measurements were made using a Nuclear Research Corp. portable neutron monitor model NP-2 (Andersson-Braun rem meter). During the neutron measurements the accelerator generated 5 Gy x-rays/min at 1 m from the head and the maximum field size was 40 cm x 40 cm at 1 m (James, 1988). The measured neutron dose rate exceeded the instrument threshold of 0.2 uSv/h at only one location. At floor level just outside the closed door of the accelerator room the neutron dose rate was 3.5 uSv/h.

O'Brien, 1985, reported the results of neutron measurements outside the concrete shielded room housing a Therac-25 25 MeV linear accelerator at the Toronto-Bayview clinic of the Ontario Cancer Foundation. The measurements were made with an Andersson-Braun type rem meter (Type NP-2, Nuclear Research Corp.) which was calibrated at NRC. the maximum dose equivalent rate at the doorway was 0.85 mSv/h with the collimator jaws fully closed, a peak photon doserate of 500 cGy/min and a gantry orientation of 90° (gantry head closest to the maze opening). The value with the jaws wide open was about 60% of the maximum value with the jaws closed. The construction of a hydraulically driven rolling door (consisting of 12 mm steel, 50 mm of lead, 127 mm of polyethylene and 5 mm of Flexboron) reduced neutron dose-equivalent rates to 0.6% (i.e. <0.5 mrem/h) of their original value.

A schematic representation of the neutron energy spectra found in occupationally accessible areas around medical electron accelerators is shown in Figure 5.1-1. All neutron energies are represented from thermal up to almost 10 MeV based on the spectra shown in Figure 2.1-1.

2.1.2 High Energy Accelerators

The Foster Radiation Laboratory at McGill University operates a K=100 Synchrocyclotron that can accelerate protons and ^{4}He to 100 MeV. A pulsed beam is produced with a pulse length of 10 us and a period of 2.5 ms. Superimposed upon each pulse is a microstructure at the RF extraction frequencies of 11.9 to 22.1 MHz. Neutron dose rates have been measured in occupationally accessible areas using an Eberline NRD-1 instrument with a BF_3 detector (Leake type rem meter). To reach these areas, the neutrons produced at the accelerator must pass through poured concrete, earth and stacked cement blocks. The highest dose rate, 30 uSv/h, was measured with a proton beam accelerated to 100 MeV and beam current of 600 nA (Nikkinen, 1988).

An upgraded model AN/S-2000 Van de Graaff positive ion accelerator is operated at the University of Western Ontario. Beams of protons, deuterons and ^{3}He and ^{4}He can be accelerated to 2.5 MeV and 20 uA current. Neutron dose rates have been measured in occupationally accessible areas using an Eberline NRD-1 rem meter (Leake-type) with a BF_3 detector. Dose rates up to 0.86 mSv/h were measured with a deuteron beam accelerated to 2 MeV at 100 nA. Personal neutron dosimetry is accomplished with the use of CR-39 track etch detectors provided by DNHW. Quarterly reported doses ranged from 0.2 to 1.2 mSv. Neutron spectra at the Faraday cup in the 32° beam line and beside the collimating slits of the 15° beam line were measured by Ing (1981) while the accelerator was operating at Chalk River Nuclear Laboratory. These locations and conditions were chosen because they gave the highest neutron fields in practice. Measurements were made using proportional counter spectrometers containing hydrogenous fillings and a ^{3}He counter. Deuterons at 2 MeV and 20 nA were directed down the 32° beam line. No neutrons were detected above 2 MeV. There were few neutrons below 0.1 MeV and the thermal neutron contribution was negligible (Figure 2.1-3). When deuterons at 950 keV were directed down the 15° beam line, neutrons were measured in the energy interval 0.2 to 0.7 MeV and there were very few less than 0.1 MeV. Again, the thermal neutron contribution was negligible (Figure 2.1-4).

The operation of the proportional counters, used by Ing (1981) to measure the neutron spectra described above, is explained in Section 3.2.3. From the proportional counter output, the energy spectra of both gamma rays and neutrons can be determined. Gamma rays produce lineal energies in the range 10^{-2} to 20 keV/um and neutrons produce lineal energies in the range 20 to 1000 keV/um (Figure 2.1-2).

The Saskatchewan Accelerator Laboratory (SAL) facility is capable of accelerating an electron beam up to 300 MeV at 45 kW of beam power. Current research does not require deliberate production of neutrons. However, neutrons are produced as secondary radiation when the beam dissipates in the beam line components and beam stops. The machine is capable of producing 2×10^{12} n $s^{-1}kW^{-1}$. The beam pulse repetition rate is variable from 1 to 360 pulses per second with an electron beam pulse length of 0.1 to 1.5 us (Choubal, 1988).

Although access to the accelerator is restricted during operation by the use of interlocked doors, personnel radiation exposures can result from neutron leakage through the primary shield. Until 1980, personnel were monitored using the DNHW neutron film dosimetry service. Records over 15 years of operation showed no measurable exposure on any of the film dosimeters and the service was discontinued (Choubal, 1988).

A recent upgrade to include an energy compression system has increased the potential for neutron radiation exposures in occupied areas. Although some neutrons with energies up to the maximum beam energy (300 MeV) will be generated, the principal component will be produced by the giant resonance processes with neutron energies less than 20 MeV. Neutron dose rates as high as 1 mSv/h have been estimated and will necessitate the addition of more shielding (Choubal, 1988). The neutron energy spectrum produced by 300 MeV electrons incident upon a lead target at SAL would be similar to that shown for 266 MeV electrons shown in Figure 2.1-5. The neutron energy spectrum in accessible areas will be somewhat degraded in energy depending upon the thickness of shielding installed.

Atomic Energy of Canada Limited operates the Tandem Accelerator Superconducting Cyclotron (TASCC) at Chalk River. TASCC facility consists of a 13 MV Tandem accelerator injecting into a four sector Superconducting Cyclotron that is designed to accelerate all ions from lithium (up to 50 MeV/u or MeV per nucleon) to uranium (up to 10 MeV/u). The worst case to shield against is provided by high energy ^{12}C ions striking a beam stop or parts of the accelerator or beam transport system. An exception to this is the bypass line where the worst case is the transmission of 26 MeV protons. The total neutron flux produced by a beam of 50 MeV/u ^{12}C ions stopping in iron was calculated and are shown in Figure 2.1-6. Neutron doses received by personnel over a one year period ranged up to 0.79 mSv.

The TRIUMF cyclotron located at the University of British Columbia is designed to accelerate negatively charged hydrogen ions to 520 MeV. The energy of the extracted proton beam is continuously variable from 180 to 520 MeV and the macroscopic duty factor of the beam can be adjusted as high as 100%. It can be varied continuously down to 1% and is achieved at a frequency of 1 kHz. For instance, in the 10% mode the beam would be present 100 us every 1 ms. The beam also has a much finer temporal structure. A 23 MHz buncher pulses the beam at 23 million proton bursts per second (one every 43 ns) and each burst lasts for 5 ns.

Neutron spectra were measured (Moritz, 1989) in accessible areas using a set of moderating spheres surrounding a 4 mm x 4 mm ^{6}LiI(Eu) scintillator. The energy response of the system is altered by using 0.96 g/cm^3 polyethylene moderator spheres of various radii. As the Bonner Sphere system is relatively insensitive to neutrons with energy greater than 10 MeV, additional measurements were made using the $^{12}C(n,2n)^{11}C$ reaction in a 50 mm diameter by 50 mm high cylinder of NE102A plastic (see Section 3.1.2).

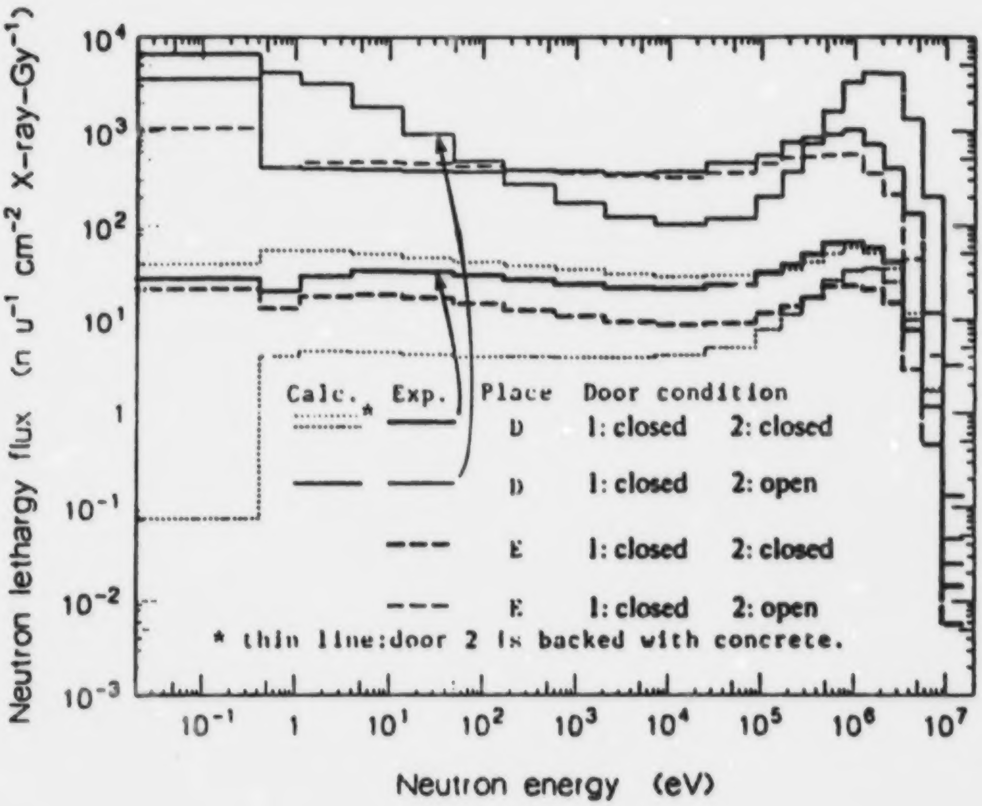
The results of the measurements are shown in Figure 2.1-7. The neutron spectrum measured at the door to the personnel entrance maze to the cyclotron vault, Figure (a), is dominated by thermal and low energy neutrons. The cyclotron is shielded locally with concrete so that the maze does not view any radiation source directly. The neutron spectrum measured at the door to the service cave on top of the high intensity beam dump is shown in Figure (b). The concrete shielded maze is very short and as a result there is a substantial high energy component in the neutron spectrum. The neutron spectrum measured beside the liquid deuterium target is shown in Figure (c). As the shield is composed almost entirely of steel, there is substantial neutron leakage near 200 keV and 100 MeV and significant low energy neutrons. The results of measurements in the biomedical facility are shown in Figure (d). Shielding consists of approximately 2 m of steel followed by approximately 4 m of heavy aggregate concrete between the meson production target and the measurement point. As at the steel shield for the liquid deuterium tank, there are a substantial number of neutrons about 200 keV and 100 MeV although in this case thermal and low energy neutrons predominate.

The results of measurements above the beam dump of the high intensity proton line are shown in Figure (e). The shielding consists of iron, water and some concrete and therefore, the spectrum is predominated by high energy neutrons.

The results of these measurements illustrate the requirement for a flat dose equivalent energy response to neutrons with energy from thermal up to about 500 MeV for both neutron survey meters and personal dosimeters used at the facility. The representation in Figure 5.1-1 of the neutron energies found in accessible areas near the 500 MeV accelerator includes all energies in the interval thermal to 500 MeV.

Figure 2.1-1

NEUTRON SPECTRA OUTSIDE SHIELDING DOORS OF MM22 MICROTRON



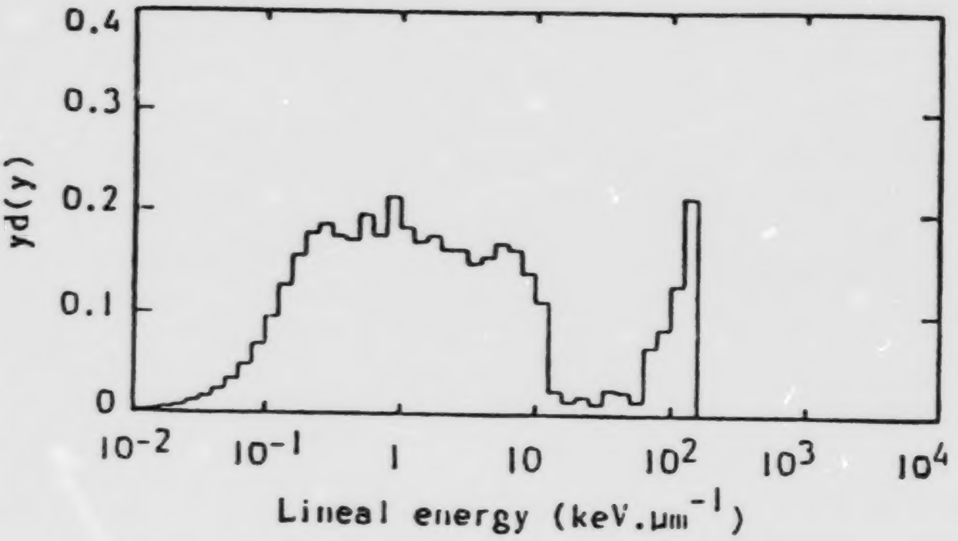
NOTE:

1. Comparison of experimental and calculated neutron lethargy spectra outside shielding doors. The experimental spectra were measured with the ^{3}He counter spectrometer. Dotted lines are calculated spectra at point D, immediately outside the outermost door. The thinnest dotted line includes the neutron backscatter effect of the concrete wall, ceiling and roof. Point E is about 2 metres removed from the outermost door and point D.

SOURCE: Uwamino, 1986

Figure 2.1-2

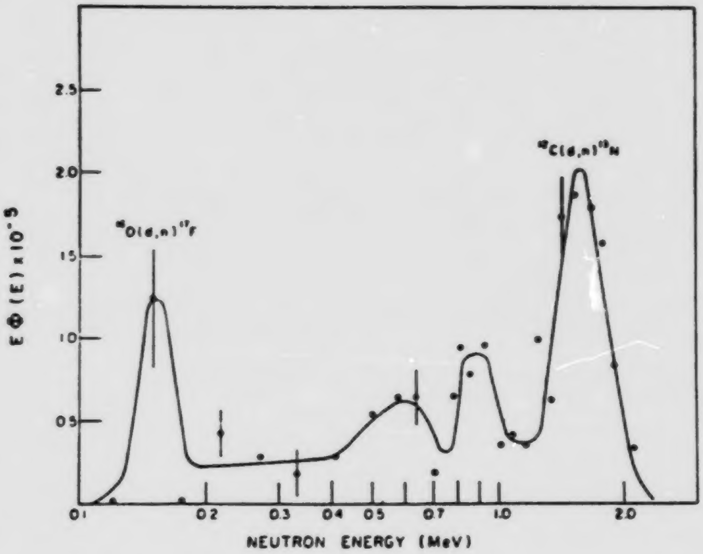
PROPORTIONAL COUNTER OUTPUT IN MIXED GAMMA AND NEUTRON FIELDS



SOURCE: Schuhmacher, 1986

Figure 2.1-3

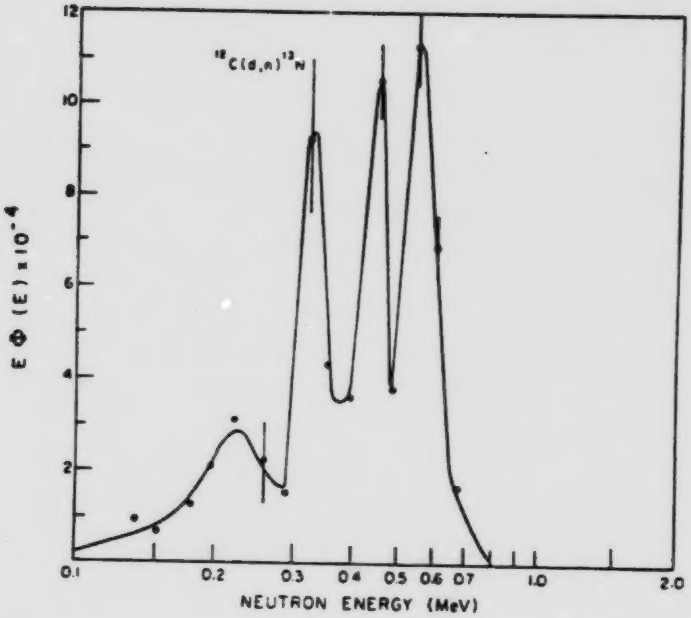
NEUTRON SPECTRUM AT 32° BEAM LINE
OF AN/S-2000 van de GRAAFF ACCELERATING
2 MeV DEUTERONS



SOURCE: Ing, 1981

Figure 2.1-4

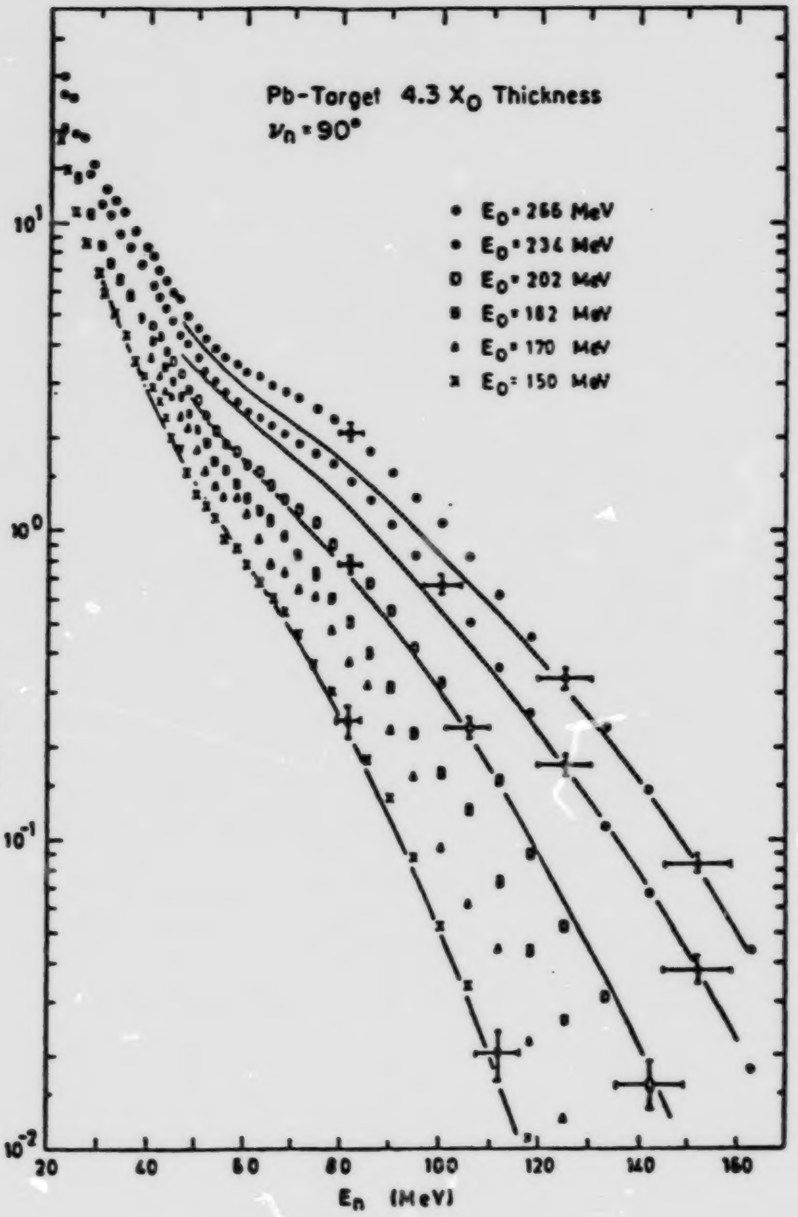
NEUTRON SPECTRUM AT 15° BEAM LINE OF AN/S-2000 van de GRAAFF ACCELERATING 950 KeV DEUTERONS



SOURCE: Ing, 1981

Figure 2.1-5

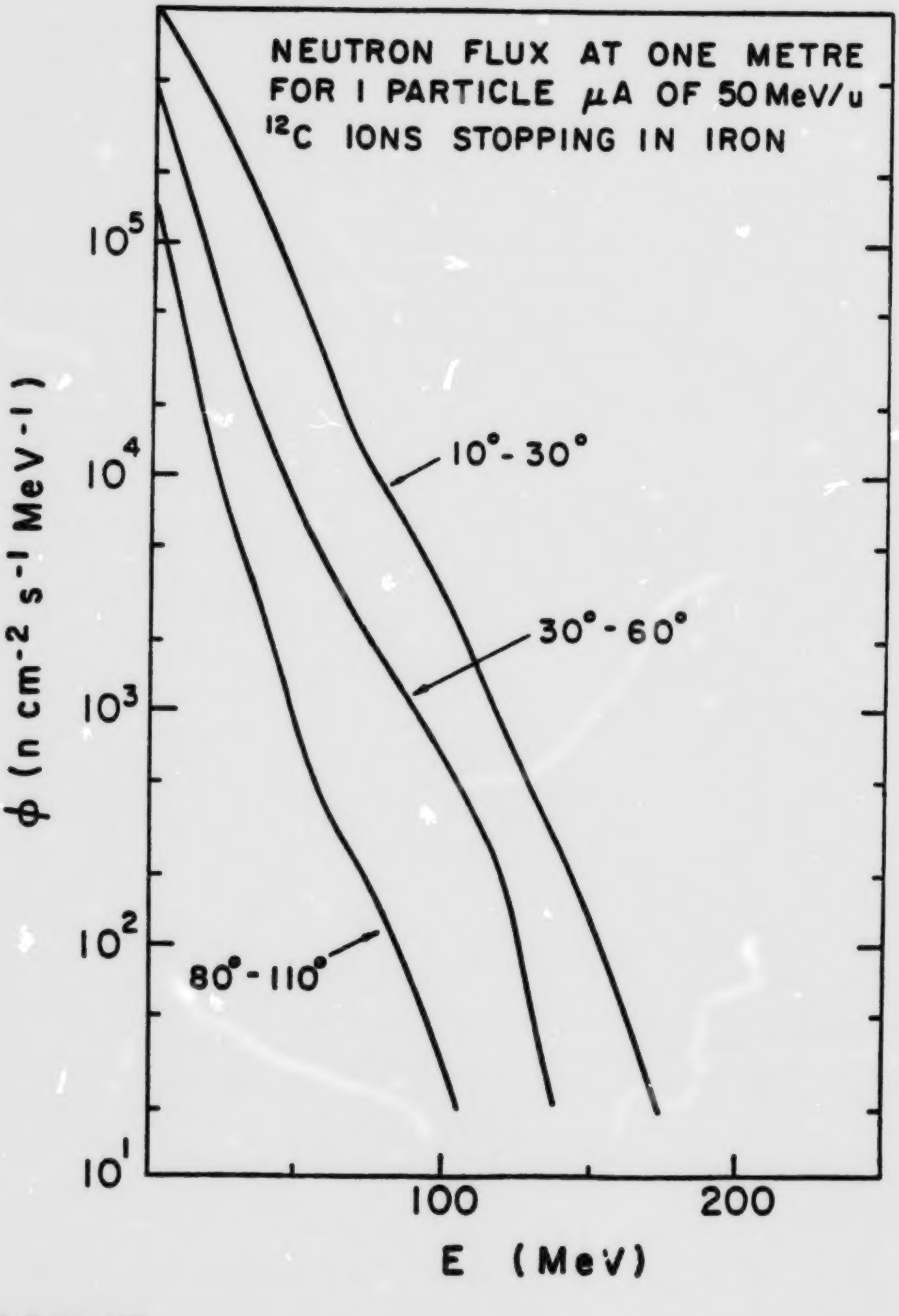
NEUTRON SPECTRA PRODUCED BY ELECTRONS OF ENERGY UP TO 266 MeV ON A LEAD TARGET



SOURCE: Choubal, 1988

Figure 2.1-6

NEUTRON SPECTRA FOR 50 MeV/μ ^{12}C IONS STOPPING IN IRON AT THE CRNL TANDEM ACCELERATOR SUPERCONDUCTING CYCLOTRON

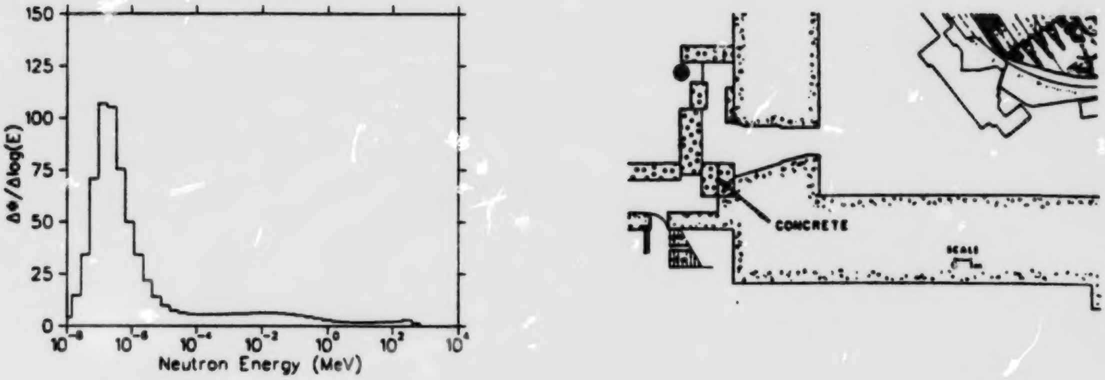


SOURCE: Bender, 1988

Figure 2.1-7

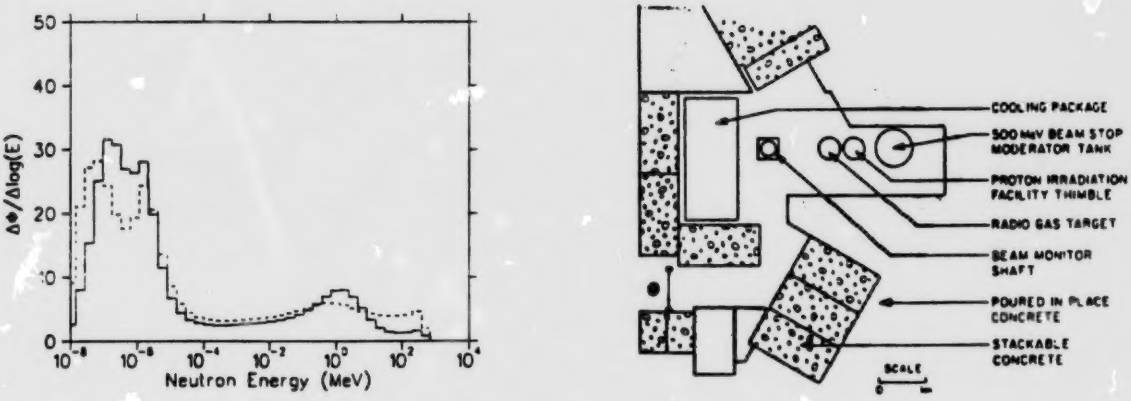
NEUTRON SPECTRA AT TRIUMF

A)



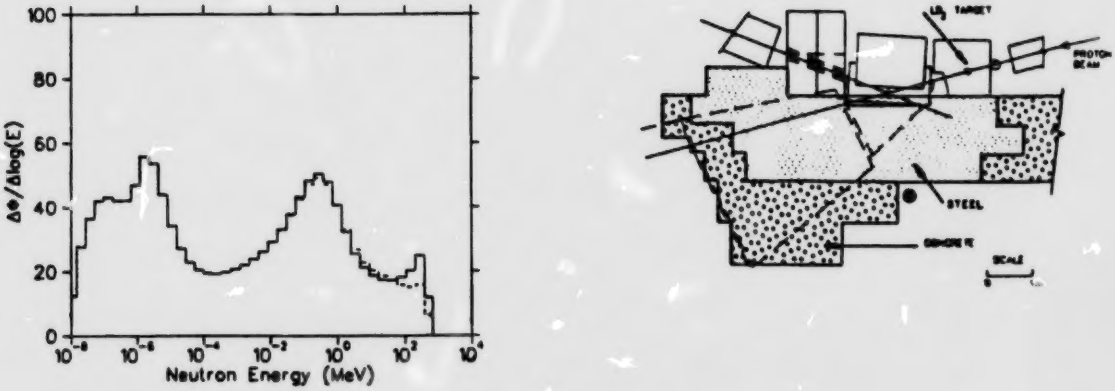
Neutron spectrum at the entrance to the 500-MeV cyclotron vault maze   Plan view of cyclotron vault entrance maze. The measurement point is indicated by the large black dot

B)



Neutron spectrum at entrance to 500-MeV beam dump service cave   Plan view of cave which is located approximately 5 m above a molten lead beamstop.

C)



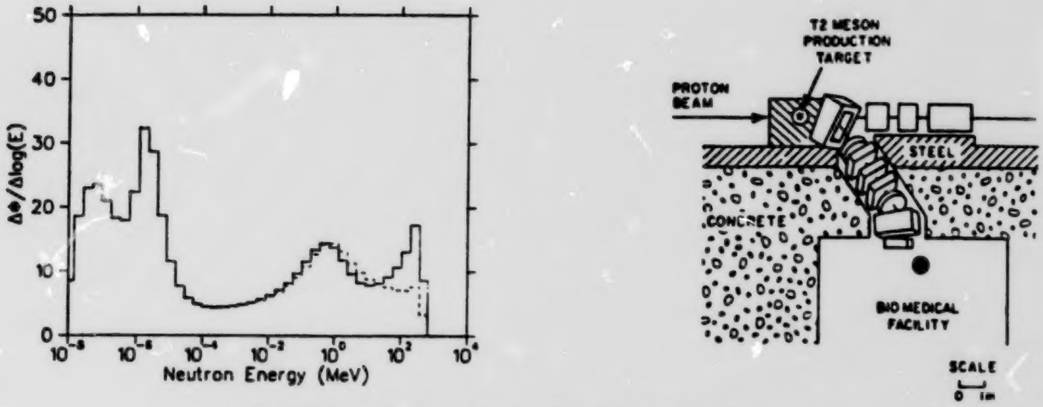
Neutron spectrum beside steel shielding of a liquid ^{2}H neutron production target.   Plan view of the liquid ^{2}H target shielding. The target is approximately 0.2 m thick and the proton beam energy is 500 MeV

SOURCE: Moritz, 1989

Figure 2.1-7

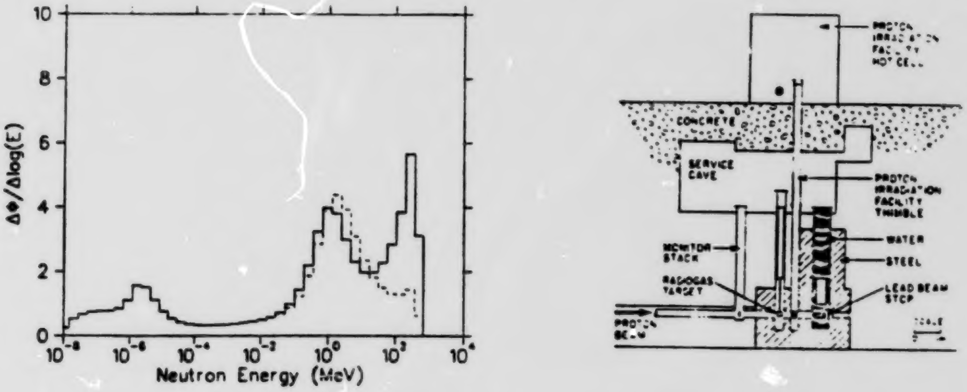
NEUTRON SPECTRA AT TRIUMF

D)



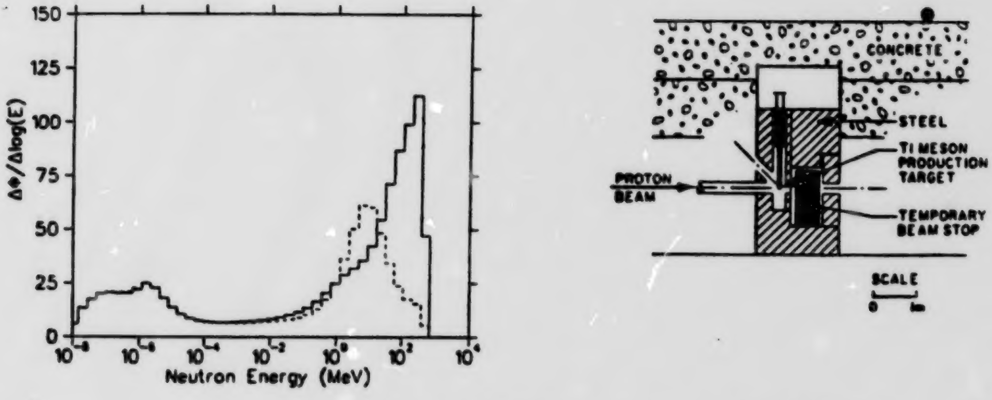
Neutron spectrum in the pion therapy facility   Plan view of the pion therapy channel shielding. The production target is 0.10 m of Be, with a proton beam of 150 μA at 500 MeV.

E)



Neutron spectrum above 500 MeV beam dump.   Vertical section through beam dump shielding. This measurement point is located above the service cave shown in Fig. 5.

F)



Neutron spectrum downstream of meson production target.   Vertical section through shielding of the meson production target. The 500-MeV proton beam was stopped in a temporary steel beam dump immediately downstream of the target.

SOURCE: Moritz, 1989

2.2 Nuclear Reactors

Dose equivalent measurements of leakage neutrons above the steel top plates of the DIDO and PLUTO reactors at Harwell were measured (Birch, 1985) using a multisphere spectrometer. Both reactors produce 25 MW thermal power and are heavy water moderated. It is thought that the spectra may be similar to those occurring inside containment buildings of CANDU reactors. The top steel plates are 0.355 m thick and have a matrix of plugged holes in the central area. Neutron leakage spectra were also measured using a high resolution spectrometer system based on spherical hydrogen-filled proportional counters. The spectra above the two reactors were found to be similar with nearly all of the neutrons at energies less than 700 keV and with most of the dose equivalent arising from neutrons with energies between 100-700 keV, Figures 2.2-1(a) and (b).

Stray neutron radiation fields to which workers are exposed inside containment of nuclear power reactors have also been reported by Sujak (1987) in Czechoslovakia. He reported that the important sources of neutron fields at nuclear power plants are:

- prompt neutrons from the reactor core with mean energies in the fission spectra region (about 2 MeV) which are degraded in energy by the moderator, reflector and biological shield;
- delayed neutrons from induced and fission nuclides in the primary coolant such as ^{87}Br, ^{17}N and others.

Neutron fluence was measured using a multisphere spectrometer consisting of 7 spheres with diameters ranging from 50 to 305 mm and a 4 x 8 mm ^{6}LiI(Eu) scintillation detector (type SKNG NO2A fy Tesla). Typical neutron spectra measured in the reactor halls are shown in Figure 2.2-2. Details concerning location and adjacent equipment were not reported. In one location, fission spectrum neutrons are leaking out of the reactor with little moderation and there is a distinct peak in the spectrum just below 1 MeV. At other locations, the spectra exhibit much broader maxima extending 5 or 6 orders of magnitude below fission energies. One of the strongly moderated spectra (mean energy approximately 50 keV) measured on the platform of the main circulation pumps is shown in Figure 2.2-3(a). A spectrum typical of those measured in the rooms around the reactor bottom (mean energy approximately 200 keV) is shown in Figure 2.2-3(b). Fission spectrum neutrons are only slightly moderated to lower energies.

Portal (1985) has also measured neutron dose equivalent spectra in accessible areas at Pressurized Water Reactors in France using the multisphere method. Figure 2.2-4(a) shows the bimodal dose equivalent energy distribution obtained at the highest accessible level in the containment building. The most prominent peak occurs between 0.1 and 0.8 MeV, with a smaller peak at thermal energies. The quantity of interest in the accessible areas of the reactor containment is the dose equivalent rather than the neutron fluence. Therefore, to emphasize those neutron energies responsible for the most significant doses, Portal manipulated the data in the following manner before plotting it on the vertical axes of the graph. The dose equivalent as a function of neutron energy, h(E), (Portal did not identify the source) was multiplied by the neutron lethargy (equivalent to the product of neutron energy and fluence, E.Q(E)), to arrive at the data h(E).E.Q(E) plotted on the vertical axes. At the lowest level in containment, Figure 2.2-4(b), the thermal peak is larger than the higher energy peak. A similar bimodal spectral shape was measured at a personnel entry lock, Figure 2.2-4(c). In the vicinity of the primary coolant circuit, Portal detected very few thermal neutrons relative to the number of neutrons in the energy range

0.1 to 0.8 MeV. These observations contrast sharply with Sujak's results. However, they are made on reactors of completely different design.

Facey (1977) used the "Delta" neutron spectrometer, described in section 3, to measure the neutron spectra in accessible areas at Ontario Hydro's heavy water nuclear power reactors. The spectrum measured inside the containment vault in front of the reactor face is shown in Figure 2.2-5(a). It was made at low reactor power during commissioning of a 750 MWe reactor at Bruce Generating Station. The one feature of the spectrum is a broad peak starting at about 10 eV, reaching a peak at 0.1 to 1.0 MeV and then quickly dropping off at higher energies.

Two spectra measured in the Pickering boiler room are shown in Figures 2.2-5(b) and (c). At one location the maximum has shifted to a lower energy, 30 to 100 keV. However, the thermal neutron contribution remains insignificant.

In the spectrum measured in the Pickering fueling machine service room, Figure 2.2-5(d), there is a bimodal structure with significant numbers of neutrons in the energy intervals 10 to 30 eV and 0.01 to 1 MeV. Two additional measurements in the boiler room at Pickering, Figure 2.2-6, illustrate two distinctly different neutron spectra. At the boiler room pump there is a broad predominant neutron peak in the energy interval 0.1 to 100 keV. Whereas, beside the boilers the neutron spectrum is obviously bimodal with a narrow peak in the energy interval 10 to 30 eV and a low broad peak from 10 keV to 3 MeV.

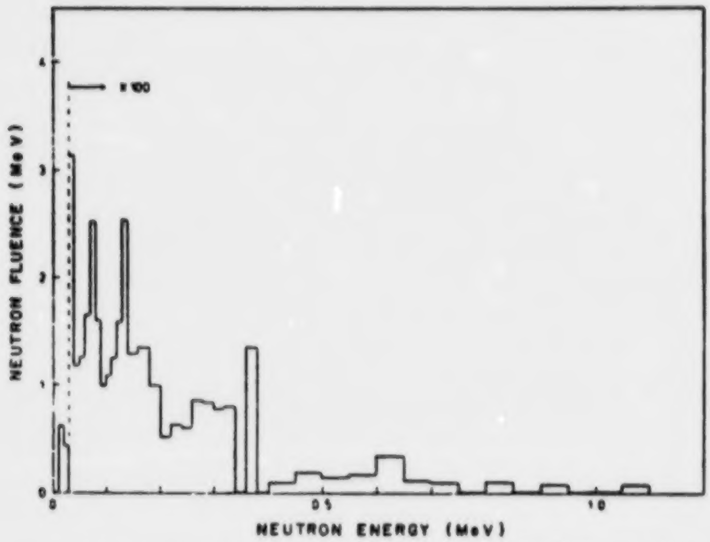
In summary, the measurements of neutron spectra at occupationally accessible points at nuclear reactors indicate the necessity of using neutron survey instruments and personal dosimeters that have a flat dose equivalent energy response from 1 keV to almost

20 MeV. It was also shown that there can be a significant neutron fluence component in the interval from thermal to about 30 eV, although the dose equivalent factor is much smaller at thermal than at higher energies. Also, it was shown that the average neutron energy can be very different at two points in close proximity in the same room. This can result in increased uncertainty in dose estimates if a spectral shape specific calibration factor must be used to interpret survey meter or personal dosimeter readings.

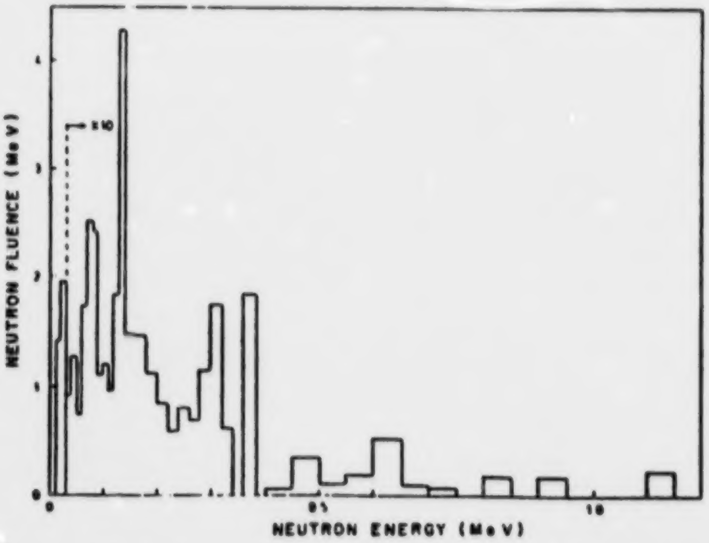
In Figure 5.1-1, the neutron energies expected to be found in accessible areas near CANDU reactors are schematically represented from thermal up to almost 20 MeV.

Figure 2.2-1

NEUTRON LEAKAGE SPECTRA THROUGH TOP STEEL PLATES OF HEAVY WATER REACTORS



A) Neutron leakage spectrum above DIDO top plate.

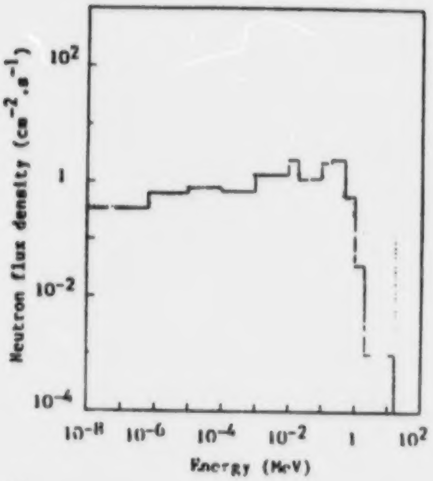


B) Neutron leakage spectrum above PLUTO top plate.

SOURCE: Birch, 1985

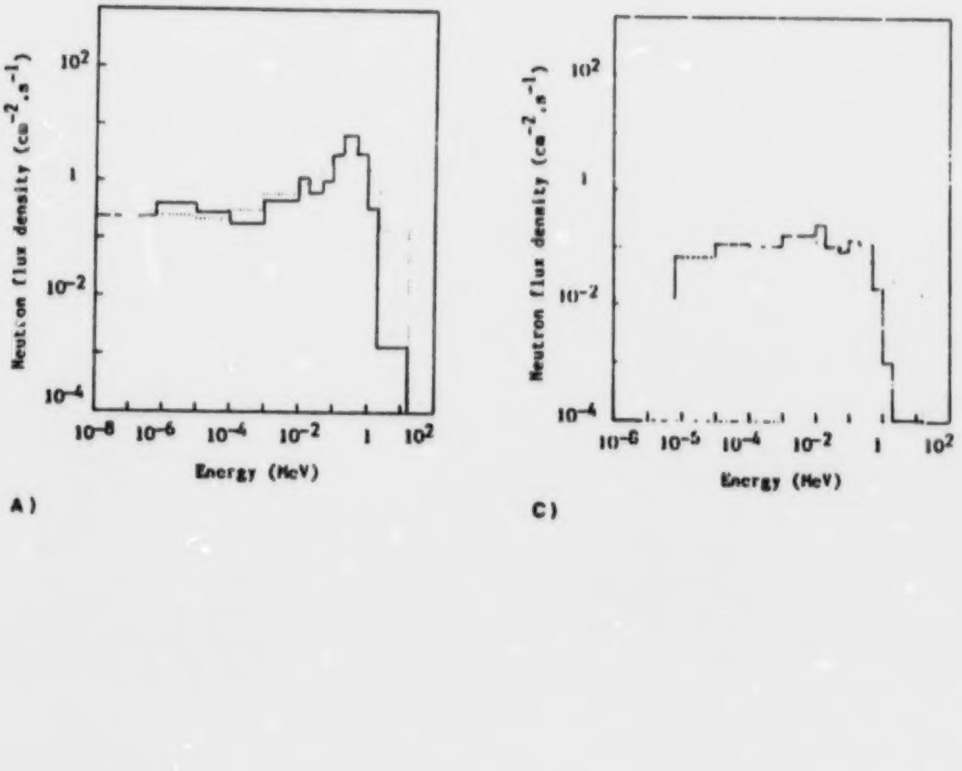
Figure 2.2-2

NEUTRON ENERGY SPECTRA IN THE REACTOR HALLS OF PRESSURIZED WATER NUCLEAR POWER REACTORS



B)

Figs. A, B and C Neutron energy spectra with mean neutron energy $\bar{E}$ and total neutron flux density φ at the reactor halls. $\varphi(E)/\ln(E_{i+1}/E_i)$.



A)

C)

SOURCE: Sujak, 1987

NEUTRON ENERGY SPECTRA
AT PRESSURIZED WATER NUCLEAR POWER REACTORS

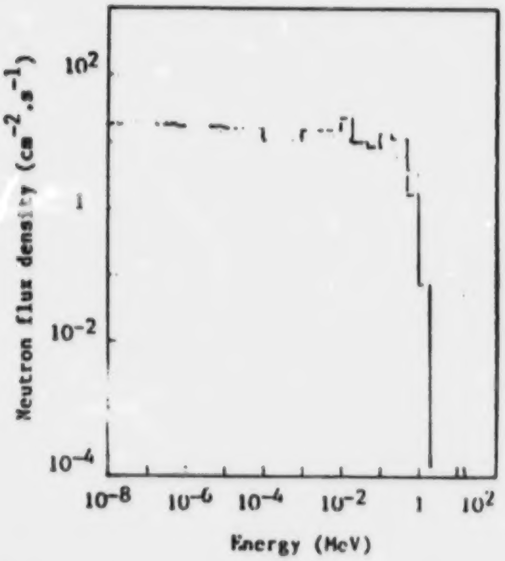


Figure A Typical neutron energy spectrum at the main circulation pump room.

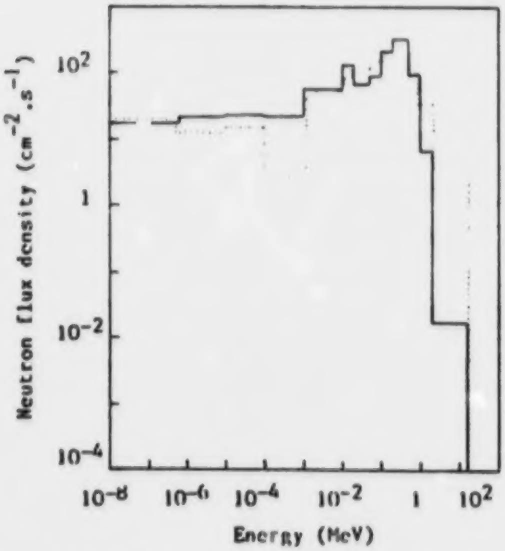


Figure B Typical neutron energy spectrum at the rooms near the bottom of the reactor.

SOURCE: Sujak, 1987

Figure 2.2-4

NEUTRON SPECTRA
MEASURED AT PRESSURIZED WATER REACTOR
(FRANCE)

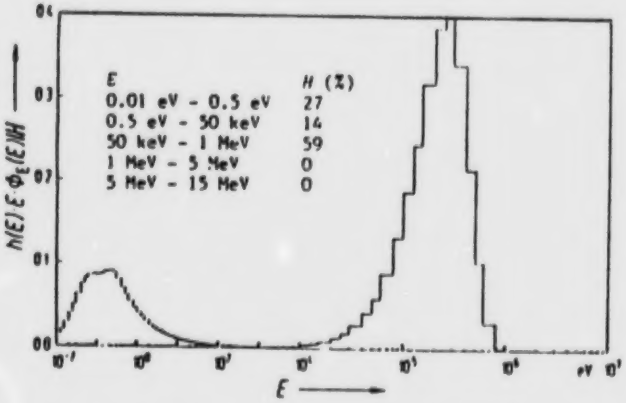


Figure A PWR, highest level. Effective energy 0.145 MeV.

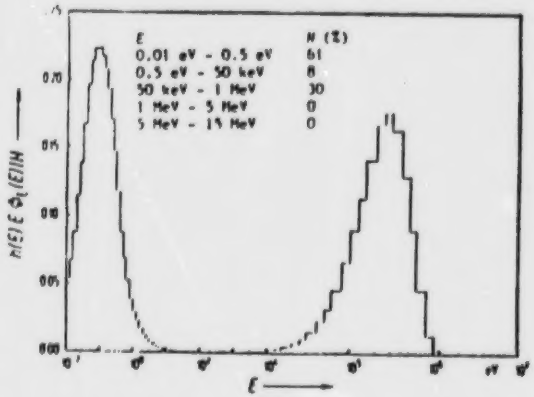


Figure B PWR, lowest level. Effective energy 0.084 MeV.

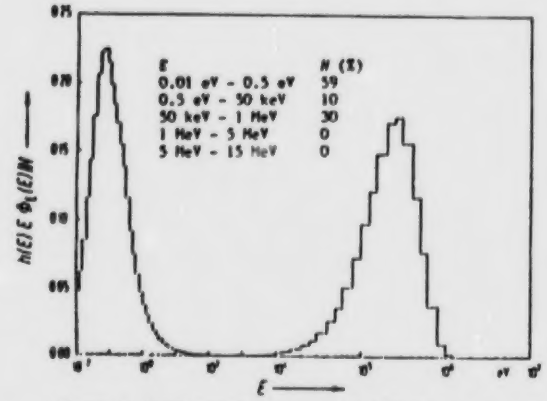


Figure C PWR, personnel entry lock. Effective energy 0.079 MeV.

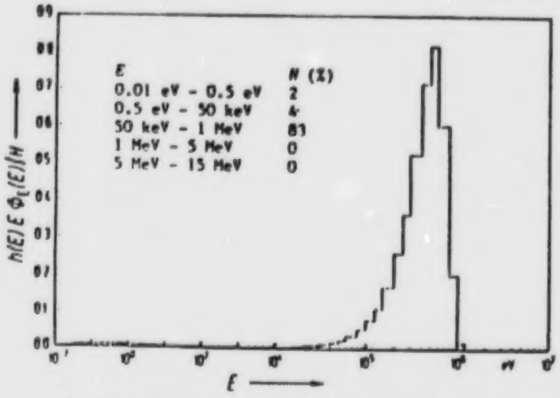
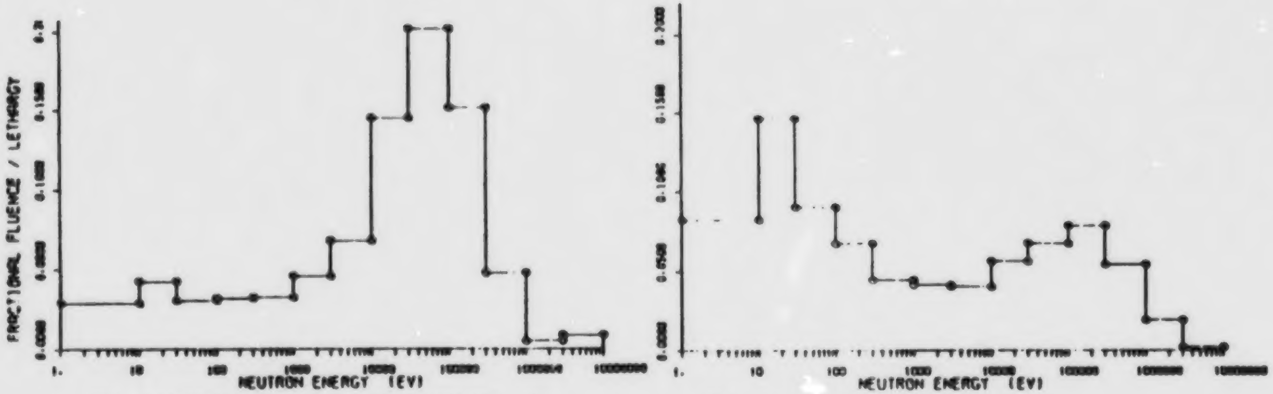


Figure D Primary water cooling circuit. Effective energy 0.415 MeV

SOURCE: Portal, 1985

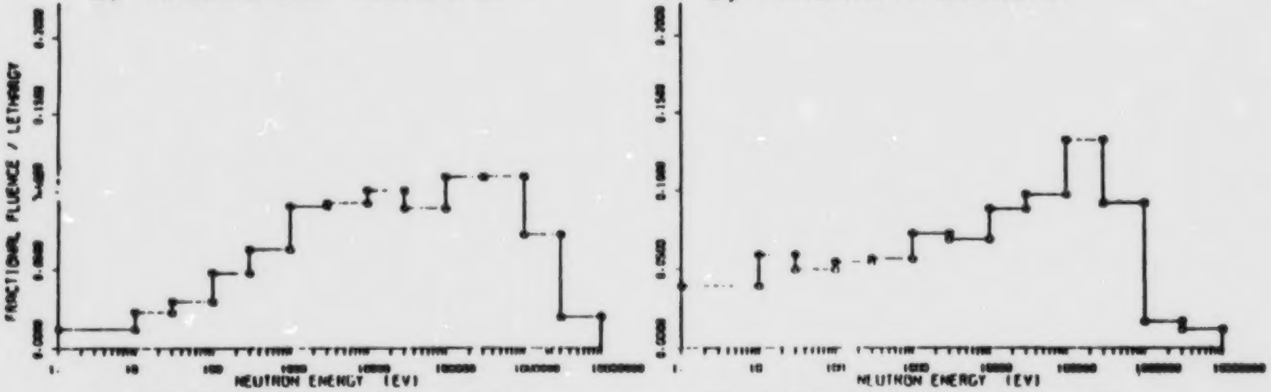
Figure 2.2-5

NEUTRON SPECTRA MEASURED AT HEAVY WATER MODERATED POWER REACTORS (ONTARIO HYDRO)



C) PICKERING B.RM. LOCATION NO. 4

D) PICKERING FM SERVICE RM



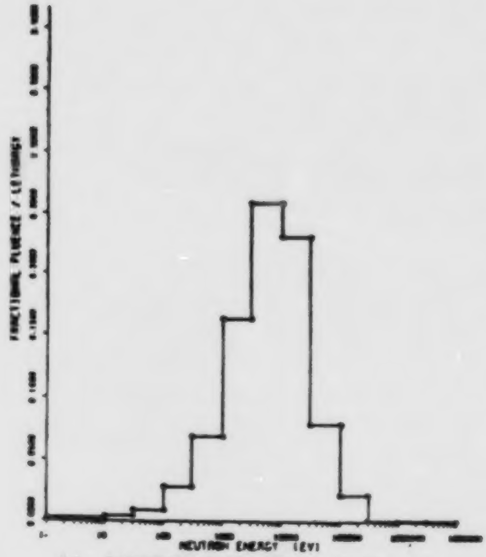
A) BRUCE-A UNIT-2 REACTOR VAULT

B) PICKERING B.RM. LOCATION NO. 1

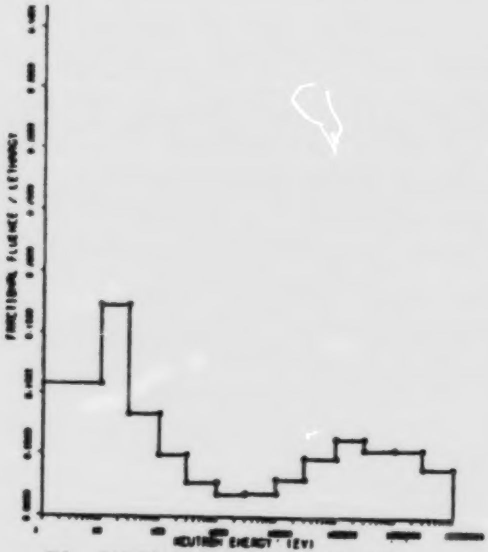
SOURCE: Facey, 1977

Figure 2.2-6

NEUTRON SPECTRA MEASURED AT HEAVY WATER MODERATED POWER REACTORS (ONTARIO HYDRO)



A) PICKERING B.RM. PRESS. PUMP



B) PICKERING B.RM. BESIDE BOILERS

SOURCE: Facey, 1977

2.3 Radioisotope Neutron Sources

Because of their stable output and relatively long half-lives, physically small sources that use (alpha, neutron) reactions or spontaneous fission as a source of neutrons are widely used in the coal and oil industries, radiography, borehole logging for water and mineral exploration and for calibrating neutron detectors and spectrometers.

The Be (alpha, neutron) sources produce the most neutrons for a given alpha activity and for many years ^{226}Ra-Be and ^{239}Pu-Be provided spectral standards for calibration. The latter combination has the advantage that the dose-rate ratio of gammas to neutrons is more than 1000 times lower than that for ^{226}Ra-Be (Table 2.3-1).

These have largely been replaced by ^{241}Am-Be sources, which combine an adequately long half-life, a reasonably high specific activity and low gamma ray emission. The gamma-ray dose rate at 1 m from a typical commercial ^{241}Am-Be source is about 10 uSv/h for a neutron source strength of 10^6 n/s. This is about 60 times less than the gamma-ray dose rate from a ^{226}Ra-Be source of the same neutron output (Cross, 1987a).

There is considerable variation in the neutron energy spectra of Be (alpha, neutron) sources depending upon the energy of the alpha particle decay and upon the grain size of the alpha emitting compound. Calculated variations in neutron spectra from Be and various alpha emitters are shown in Figure 2.3-1 (Cross, 1987a). The neutron production rate in the ranges of 3 to 7 MeV and 0.1 to 1.5 MeV are sensitive to the alpha source (i.e. alpha energy). The neutron energy maximum only varies in the range 10.5 to 11.5 MeV and is relatively unaffected by alpha energy.

The neutron energy spectrum from ^{241}Am-Be has been measured and is shown in Figure 2.3-2 (Cross, 1987a) . Neutrons are produced from about 100 keV to a maximum of 11 MeV.

Radioisotopic sources comprising B(alpha, neutron) and Li(alpha, neutron) produce neutrons of considerably lower average and maximum energy than the Be (alpha, neutron) reactions. Alpha reactions on boron produce neutrons ranging in energy from about 1 to 5 MeV, with an average of about 2.5 MeV (Figure 2.3-3(a)). Whereas, alpha reactions on lithium result in neutrons ranging in energy from about 100 keV to 1.5 MeV, with a maximum at about 200 keV (Figure 2.3-3(b)).

^{252}Cf is a spontaneous fission neutron emitter with a half-life of 2.64 years which has found widespread use in research and hospital settings. The unmoderated neutron spectrum has a Maxwellian form and a maximum energy of about 20 MeV (Figure 2.3-4). The large numbers of neutrons with approximately thermal energies are mainly products of neutron moderation within the experiment room. Delayed neutrons emitted by fission products follow the fission process but their intensity is too low to distort the prompt fission spectrum significantly.

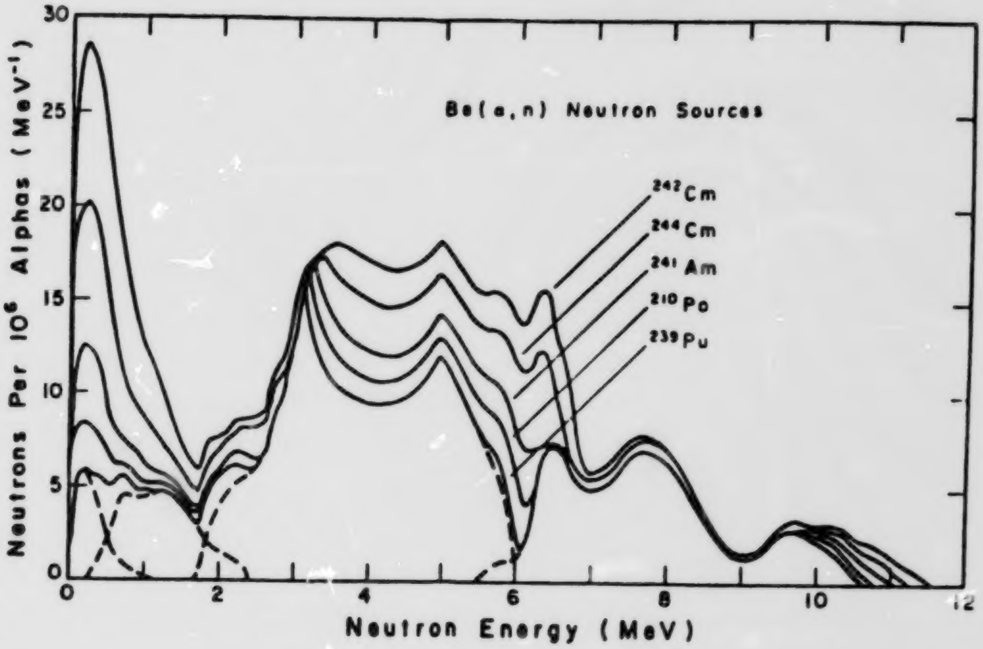
Table 2.3-1

GAMMA DOSE RATE AT 1 M FROM NEUTRON SOURCES PRODUCING 10^6 N/S

^{226}Ra-Be	600 uSv/h
^{239}Pu-Be	<0.6 uSv/h
^{241}Am-Be	10 uSv/h

Figure 2.3-1

CALCULATED SHAPES OF NEUTRON SPECTRA FROM Be (α, n) REACTIONS FOR VARIOUS SOURCES OF ALPHA PARTICLES



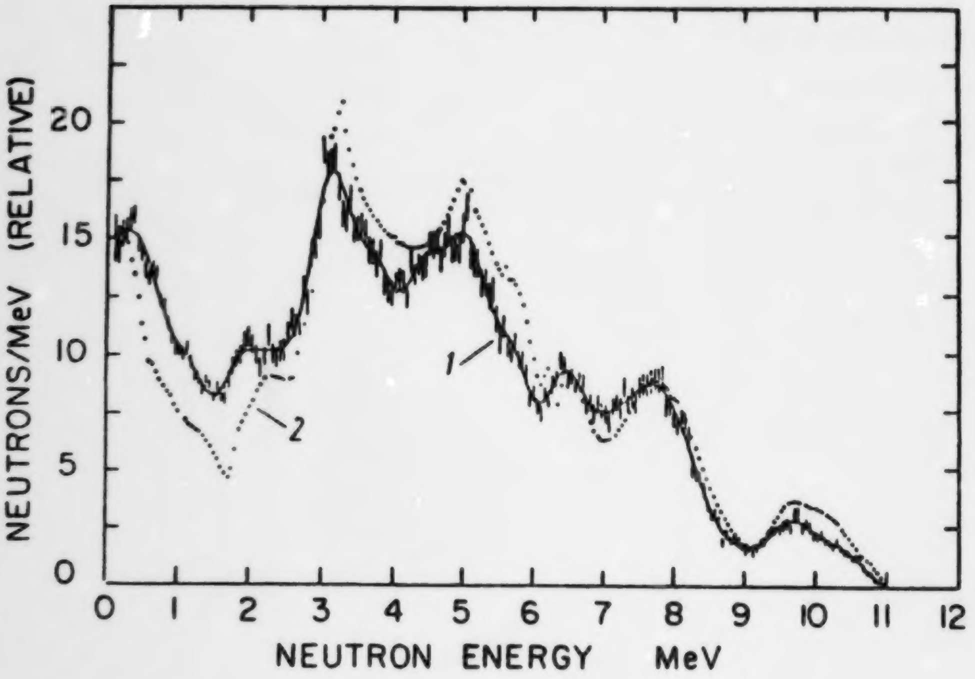
NOTE:

1. The dashed curves show individual neutron groups for the ^{239}Pu-Be source

SOURCE: Cross, 1987a in "The Dosimetry of Ionizing Radiation" vol. 2 ed. Kase et al

Figure 2.3-2

NEUTRON SPECTRUM FROM AN ^{241}Am-Be SOURCE



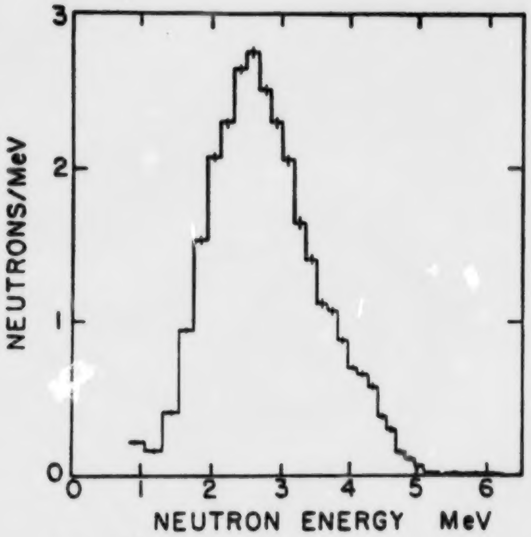
NOTE:

1. The full curve (1) shows the measured spectrum
2. The dotted curve (2) is a calculated spectrum

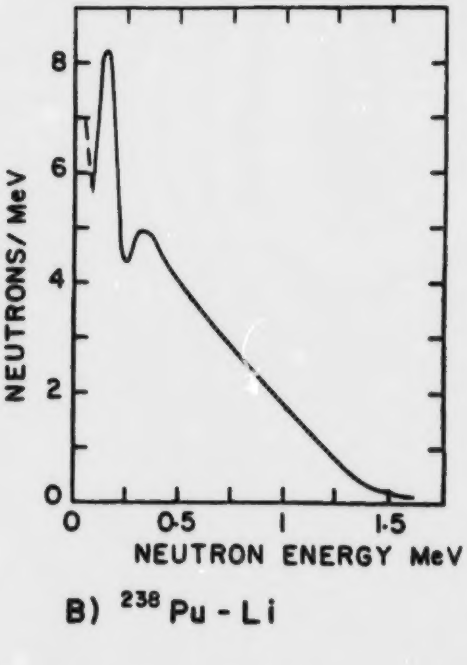
SOURCE: Cross, 1987a in "The Dosimetry of Ionizing Radiation"
vol. 2. ed. Kase et al

Figure 2.3-3

SPECTRA FROM ^{241}Am-B (α,n) MEASURED WITH A STILBENE SCINTILLATOR AND ^{238}Pu-Li (α,n) MEASURED WITH FOUR HYDROGENOUS PROPORTIONAL COUNTERS



A) ^{241}Am - B

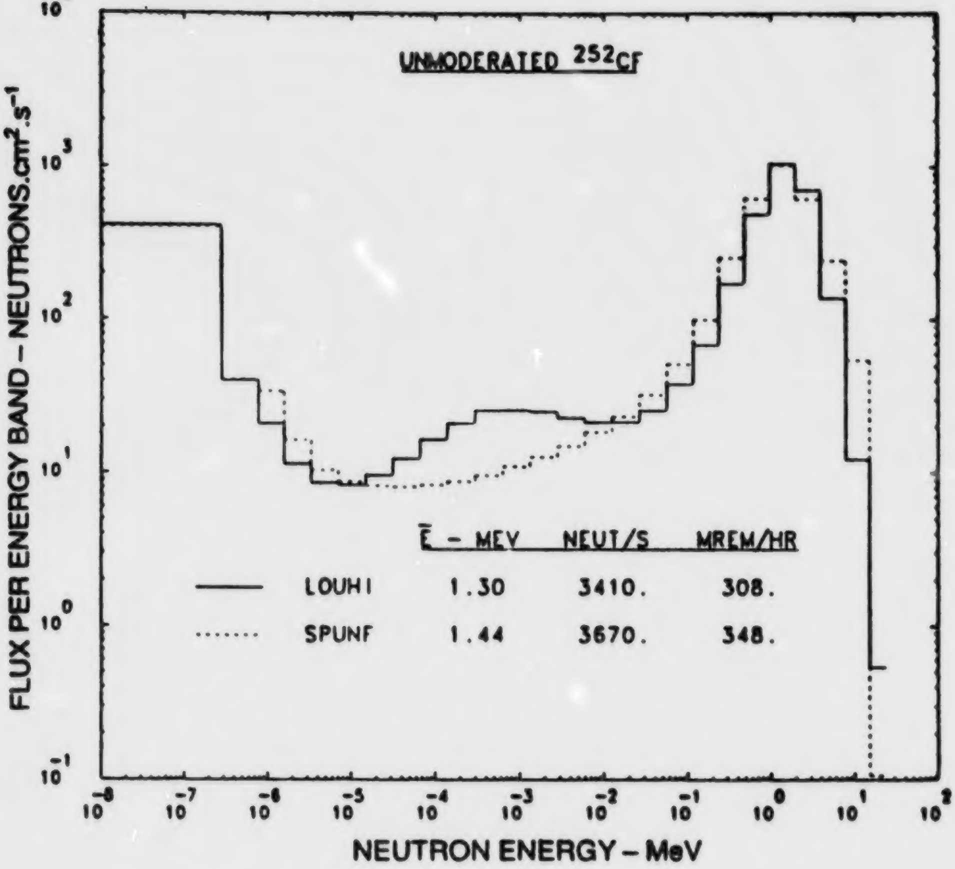


B) ^{238}Pu - Li

SOURCE: Cross, 1987a "The Dosimetry of Ionizing Radiation" vol. 2. ed. Kase et al

Figure 2.3-4

NEUTRON SPECTRUM FROM ^{252}CF TAKEN WITH MULTISPHERE SPECTROMETER



NOTE:

1. LOUHI and SPUNF are two different unfolding techniques

SOURCE: Thorngate, 1985

3.0 PORTABLE AND FIXED NEUTRON MONITORS

3.1 Activation Detectors

The application of activation detectors to measure neutron fluence (n/cm^2) involves the use of a material that becomes radioactive when exposed to neutrons. After a measured exposure time the activation detector is removed to a calibrated beta or gamma measuring facility for counting. The radioactivity induced during neutron exposure may result from capture reactions, inelastic scattering leading to an isomeric state, (neutron, proton), (neutron,alpha) or (neutron,2 neutron) reactions, induced fission or spallation reactions.

There are several advantages to the use of activation detectors. Because they are relatively inexpensive and can be reused, many detectors can be deployed at the same time. They are relatively small and can give good geometric resolution of neutron fields which have high spatial gradients. They are passive, immune to most environmental effects such as temperature, humidity and radiofrequency interference and can accurately integrate the cumulative fluence over pulsed or rapidly varying neutron fields (associated with accelerators for example). Likewise, some disadvantages may be encountered with activation detectors. A delay is always associated with counting the induced activity. To achieve high sensitivity, it may be necessary to use sophisticated and expensive counting equipment.

3.1.1 Thermal Neutron Activation Detectors

There are many materials, usually metallic foils, that have suitable half-lives, purities and decay emissions to be used for measuring thermal neutron fluence. Selected materials in common usage are listed in Table 3.1-1. Indium and gold are among the most common detector materials used since they are readily available at a suitable purity and provide high sensitivity because of their high neutron absorption cross-section.

More thermal neutron detectors produce radioactive nuclides under neutron irradiation. In these cases, neutron fluence is calculated from the absorption cross-section, the measured absolute activity of the detector and the corrections for irradiation, decay and counting times. However, some of the thermal neutron detectors listed in Table 3.1-1 are stable isotopes. In these cases, capture-gamma radiation produced during neutron irradiation is measured or the detector is a component of a gas in an ionization chamber detector.

Some thermal neutron detectors have finite sensitivities to higher energy neutrons. To evaluate the activity induced by truly thermal neutrons a double exposure system is used. When the (bare) detector is exposed, radioactivity is produced by both thermal and higher energy neutrons. A second irradiation is made of the same detector material wrapped in a cover of a substance such as cadmium which is effectively opaque to thermal neutrons. The radioactivity produced in the second detector is due only to neutrons of energy higher than the cadmium cut-off. Subtraction of the activity produced in the cadmium-covered detector from that produced in the bare detector yields the activity produced by thermal neutrons only.

3.1.2 Threshold Activation Detectors

Threshold activation detectors facilitate spectroscopy in the neutron energy range about 0.7 to 12 MeV. They are characterized by a neutron activation cross section that is insignificantly small at low neutron energies and then rises steeply at a unique threshold energy. Above the threshold, the activation cross section is typically broad and extends to much higher energies. Half-lives of threshold detectors are usually the order of hours with some as long as days and others as short as minutes (Table 3.1-2). Estimates of spectral structure in the neutron energy range 0.7 to 12 MeV can be made by mathematically manipulating or unfolding results of the irradiation and counting of detectors of selected activation threshold energies. The spectral resolution achieved is directly proportional to the number of different threshold detectors employed.

The $^{12}C(n,2n)^{11}C$ reaction is used to measure neutron fluence above about 20 MeV. ^{12}C is the natural isotope of carbon and is frequently used in the form of plastic scintillators. The plastic scintillator is placed into the neutron flux to be measured. ^{11}C is produced by the (n,2n) reaction with neutrons of energy greater than 20 MeV interacting with ^{12}C. After irradiation, the scintillator is placed onto a photomultiplier and the positrons emitted during the decay of ^{11}C are measured. The 20.34 minute half life of ^{11}C allows reasonable irradiation and counting times (one or two half lives) and permits transfer from irradiation position to counter without significant loss of activity. The cross section for the reaction is well known (Figure 3.1-1) and the neutron fluence can be calculated. The neutron energy range (above 20 MeV) over which the carbon scintillator can be used is represented schematically in Figure 5.1-1.

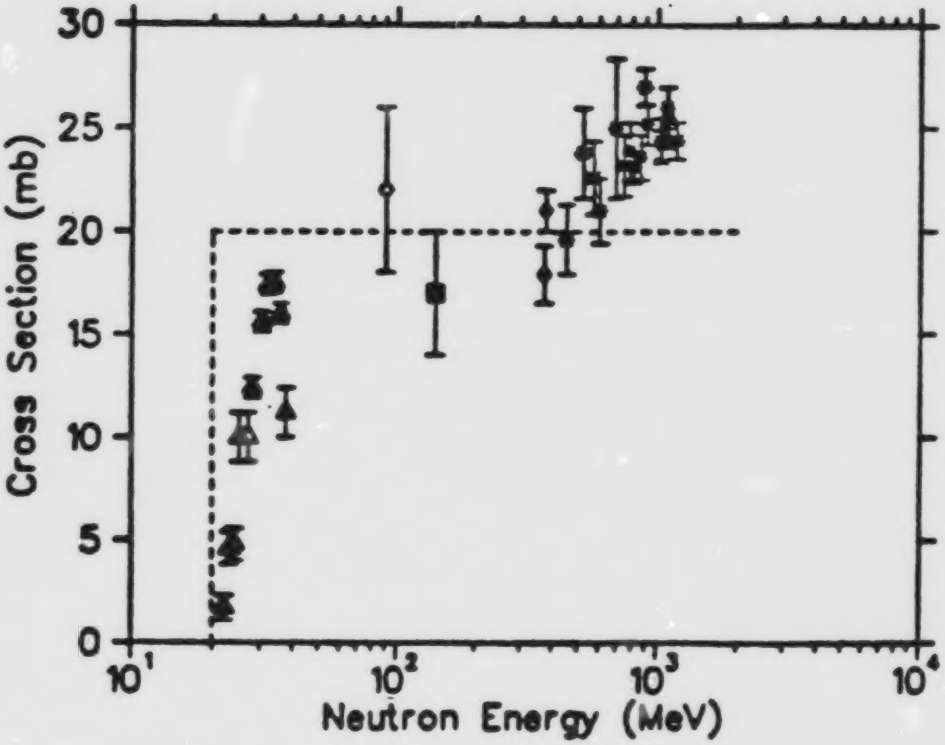
High energy photons can initiate competing nuclear reactions in the threshold detector itself which interfere with threshold activation measurements. Consequently, care must be taken to ensure that high energy photons are not present or that their influence has been taken into account. Some researchers have reported success in measuring high energy neutron fluence with ^{32}P and ^{31}Si which are produced by neutron capture only and not by any photoneutron mechanisms.

3.1.3 Moderated Foil Detectors

Another method of measuring fast neutrons is to use hydrogenous material to moderate the neutrons and produce a thermal fluence inside the moderator. A thermal activation detector is placed in the center of the moderator to indirectly measure the fast neutrons incident on the surface of the moderator. The composition and shape of the moderator influences the neutron response as a function of energy. Suitable moderators are comprised mainly of hydrogenous compounds such as polyethylene, paraffin and water. The mass of the hydrogen nucleus (proton) is essentially equal to the mass of the neutron, thereby facilitating the largest energy transfer per collision. In the ideal case, the dose equivalent response of the moderator/detector system would be independent of energy. This cannot be achieved with a simple sphere or cylinder of moderator surrounding a thermal activation detector because the slowing down and absorption properties of the moderator diverge from the ICRP recommendations of dose equivalent as a function of neutron energy.

Figure 3.1-1

CROSS SECTION FOR THE ^{12}C THRESHOLD ACTIVATION DETECTOR



SOURCE: Moritz, 1987

Table 3.1-1

USEFUL THERMAL NEUTRON DETECTORS[a]

Nuclide or Element	Reaction	Half Life of Nuclide Produced	$\sigma_0 \times 10^{24}$ for Production cm^2
^{3}He	(n,p)^{3}H	12.3 y	5327
^{6}Li	(n,t)^{4}He	stable	945
^{10}B	(n,α)^{7}Li	stable	3837
^{23}Na(IS)[b]	(n,γ)^{24}Na	15.0 h	0.534
^{45}Sc(IS)[b]	(n,γ)^{46}Sc	85 d	22.3
^{51}V	(n,γ)^{52}V	3.8 min	4.9
^{55}Mn(IS)[b]	(n,γ)^{56}Mn	2.58 h	13.3
^{59}Co(IS)[b]	(n,γ)^{60}Co	5.24 y	36.6
^{63}Cu	(n,γ)^{64}Cu	12.8 h	4.5
^{115}In	(n,γ)^{116m}In	54 min	157
^{157}Gd	(n,γ)^{158}Gd	stable	242000
^{197}Au(IS)[b]	(n,γ)^{198}Au	2.70 d	98.8
^{236}U	fission	many	577

[a] Adapted from ICRU (1969).

[b] (IS) denotes natural monoisotopic element.

Table 3.1-2

FAST NEUTRON THRESHOLD REACTIONS[*]

Reaction	Half Life	Effective threshold MeV
$^{19}F(n,2n)^{18}F$	110 min	12.1
$^{24}Mg(n,p)^{24}Na$	15.0 h	7.0
$^{27}Al(n,p)^{27}Mg$	9.46 min	4.5
$^{27}Al(n,\alpha)^{24}Na$	15.0 h	7.1
$^{31}P(n,p)^{31}Si$	2.62 h	2.4
$^{32}S(n,p)^{32}P$	14.3 d	2.7
$^{46}Ti(n,p)^{46}Sc$	83.8 d	3.8
$^{54}Fe(n,p)^{54}Mn$	313 d	3.3
$^{55}Mn(n,2n)^{54}Mn$	313 d	11.5
$^{56}Fe(n,p)^{56}Mn$	2.59 h	6.1
$^{58}Ni(n,p)^{58}Co, ^{58m}Co$	71.3 d, 9.15 h	2.8
$^{58}Ni(n,2n)^{57}Ni$	36.0 h	12.3
$^{63}Cu(n,2n)^{62}Cu$	9.76 min	12.4
$^{64}Zn(n,p)^{64}Cu$	12.7 h	4.0
$^{65}Cu(n,2n)^{64}Cu$	12.7 h	11.2
$^{103}Rh(n,n')^{103m}Rh$	56.1 min	0.7
$^{107}Ag(n,2n)^{106}Ag$	24.1 min	10.7
$^{115}In(n,n')^{115m}In$	4.50 h	1.4
$^{127}I(n,2n)^{126}I$	12.8 d	11.5
$^{232}Th(n,f)$ fission products		1.4
$^{237}Np(n,f)$ fission products		0.7
$^{238}U(n,f)$ fission products		1.4

---

[*] Adapted from ICRU, 1977.

3.2 Neutron Dose Equivalent Meters

An early design for a portable neutron survey meter was based upon the principle of thermalization of neutrons in a moderator and subsequent registration in a centrally located thermal neutron detector. Various moderating sphere diameters were proposed with the aim of achieving a dose equivalent energy response. Some researchers suggested that the optimum simple spherical moderator would have a diameter of 30.5 cm. The measured and calculated dose equivalent energy responses of a 30.5 cm diameter sphere to a plane parallel neutron beam are shown in Figure 3.2-1. There is a low dose response for thermal neutrons; the response increases with energy to a maximum at about 10 keV and then falls off uniformly toward higher energies. The maximum over response at 10 keV is about a factor of 6 relative to a normalized response of unity at 1 MeV. The response decreases to about 0.3 at 25 MeV.

simple moderating sphere described above has an satisfactorily high dose equivalent energy response over intermediate neutron energies. Also, dose from neutrons with energies greater than 10 MeV is appreciably underestimated. The weight of such an instrument (14 kg) makes it very unwieldy as a portable meter. Improvements on this design are described below.

Early rem meters frequently incorporated LiF scintillators (enriched in ^{6}Li) into the design of the central thermal neutron detector. However, its sensitivity to gamma radiation was unacceptably high and detectors of choice are now ^{3}He and BF_3 proportional counters. Passive detectors, activation foils and nuclear track detectors are also used. They are particularly useful in pulsed neutron fields, near high-energy accelerators, for example, and for long term measurements.

3.2.1 Andersson-Braun Survey Meters

An improved dose equivalent energy response has been demonstrated by Andersson and Braun who modified the simple moderating sphere in the following way (Cosack, 1985). They were the first to position a thermal neutron absorber between two cylindrical and coaxial moderating shells surrounding a BF_3 detector (Figure 3.2-4). This configuration has the advantage of reducing the overresponse to intermediate energy neutrons compared to the response of the single moderator described in the last section. In the Andersson-Braun moderator/detector, dose equivalent response is flattened because some of the incident intermediate energy neutrons are moderated in the outer shell and removed at the absorber. Higher energy neutrons pass through the absorber as they are not thermalized until reaching the inner moderator. To maintain overall sensitivity to thermal neutrons, holes in the absorber allow thermal neutrons incident on the outer shell to reach the detector. The depth of penetration of the neutrons into the moderator depends upon their energy and the response for different energies can be altered by varying the thickness and position of the absorbing layer.

The variation in dose equivalent response to a plane parallel neutron beam has been reported by several authors (Cosack, 1985; Rogers, 1981) and the relative response is shown in Figure 3.2-5. The highest sensitivity occurs when the neutron source is located at the side of the cylindrical moderator. Depending upon neutron energy, the relative response drops to about 0.85 and 0.65, at neutron energies of 15.5 MeV and 1 MeV, respectively when the source is located at the ends of the cylinder.

The dose equivalent energy response of an Andersson-Braun rem meter relative to ICRP recommendations is shown in Figure 3.2-2. Instruments of this type are manufactured by Nuclear Enterprises

Inc. (model NM2), Alnor (model 2202) and Victoreen (model Snoopy 478). Ontario Hydro has evaluated representative samples of each of these three instruments and documented several of their characteristics (Lopez, 1988). Measured dose equivalent response to neutron energy and moderator/detector orientation were similar to those reported above. At a neutron dose rate of 0.2 mSv/h, temperature dependencies were less than 1.5%/°C. At 98% relative humidity the Alnor instrument failed (short in high voltage circuitry) and the response of the other two instruments remained unchanged from lower humidity conditions. At 0.235 mSv neutrons/h, the meter readings increased less than 0.1% when each instrument was subjected to gamma dose rates up to 0.5 Sv/h.

3.2.2 Leake Survey Meters

The next improvement in dose equivalent survey meters was devised by Leake (Cosack, 1985). He adapted the Andersson-Braun concept to a spherical moderator which included a thermal neutron absorber (Figure 3.2-6). The outer diameter was 20.8 cm and the detector was light (5 kg) and portable. The inner sphere, diameter 5.5 cm, was covered with cadmium which was perforated to provide sensitivity to thermal and low energy neutrons. A ^{3}He proportional counter was used to give improved discrimination against photon radiation over that of the BF_3.

Measurements of dose equivalent energy response relative to ICRP recommendations showed generally good agreement above 100 keV (Figure 3.2-3). There is a maximum overresponse at about 10 keV. At 27 keV the overresponse by a factor of 3 was measured in relation to the value at 1 MeV. The response to thermal neutrons is about 0.6, almost double the relative response of the Andersson-Braun type survey meter. At higher neutron energies, about 15 MeV, the response is 0.35. The energy response of the rem survey meters is represented schematically in Figure 5.1-1.

The directional response was fairly isotropic because of the spherical symmetry. However, there remained thermal neutron leakage through the void containing the connector and cable. If electronic equipment was directly mounted onto the moderator, neutron scattering and shielding considerably influenced the properties of the counter.

Leake type rem meters are manufactured by companies such as Nuclear Enterprises Inc., for example. The Nuclear Enterprises MK7 model is equipped with the ^{3}He detector, measures neutron dose equivalent rates from 10 uSv/h to 10 mSv/h and has a gamma rejection ratio of 3000 to 1.

When ^{3}He or BF$_3$ proportional counters are used in a moderator to measure dose equivalent from a pulsed source such as an accelerator, there may be concern that the meter is underresponding due to pulse pile-up. A mathematical expression to evaluate the significance of the interference can be derived from basic parameters describing the source and moderator/detector systems (Cosack, 1985). With each source pulse, neutrons impinge upon the outer surface of the moderator, diffuse through it and register in the central detector. The arrival of neutrons at the detector will be spread out in time, even from an infinitely short pulse of neutrons, because of the time required for them to diffuse through the moderator. Typical times to record half the total number of counts produced by an instantaneous pulse of neutrons are 25 us for a Leake instrument and about 100 us for an instrument with a moderator diameter of 30 cm. If successive source pulses have a separation of about these times the detector receives neutrons almost continuously. If the pulse repetition rate decreases, dead time losses are negligible as along as:

$$\frac{N.T}{f\ T_o} << 1$$

where T is the detector dead time, T_o is the diffusion time characteristic of the moderator, f is the repetition rate of the pulsed source, and N the observed counting rate. Provided these conditions hold, rem meters can be operated safely in the pulsed mode to measure pulsed neutron sources.

An example will serve to illustrate the application of this relationship. Suppose that a Leake type instrument is used to measure neutron dose equivalent near an accelerator operating at 360 pulses per second, f. The characteristic diffusion time, T_o, is 25 us and the detector dead time, T, is 150 us. The above equation can be rearranged to estimate the maximum acceptable counting rate and the appropriate values inserted to yield:

$$N << \frac{360 \times 25 \times 10^{-6}}{150 \times 10^{-6}}$$

$$N << 59.6/s$$

As long as the count rate is no greater than about 10% of this limit, 6/s, there will be no underresponse. A typical detector efficiency of 0.2 counts per second per uSv/h corresponds to 30 uS/h as an upper limit on dose equivalent rate that can be measured with confidence of negligible underresponse.

3.2.3 Tissue Equivalent Proportional Counters

The tissue equivalent proportional counter (TEPC) is a cavity chamber detector which can be used to measure neutron absorbed dose. In addition, the measured lineal energy distribution in the tissue equivalent gas inside the chamber can be used to discriminate between neutron and photon events and to evaluate the dose equivalent.

Brackenbush (1985) has presented a detailed review of the TEPC and its characteristics. Basically, the detector wall which consists of tissue equivalent plastic is sealed to contain a tissue equivalent gas and proportional counter anode. External radiation interacts with the detector wall and knocks charged particles into the gas cavity (external photons knock electrons from the wall into the gas and neutrons knock protons from the wall). The gas pressure is so low that most particles have a constant rate of energy loss in traversing the cavity. Provided that the wall thickness exceeds the range of the secondary charged particles and the gas and the wall have the same composition, then the dose in the cavity is the same as the dose in the solid tissue-equivalent material. The absorbed dose can be estimated directly from the energy deposited and the mass of the particles.

It has been demonstrated (Brackenbush, 1985) that, in practice, a micro-processor based instrument can be used to analyse the height of each signal from the TEPC and sum over all energies to determine the absorbed dose. It is possible to measure neutron and photon doses independently because the lineal energy deposited by photon initiated events have distinctly lower pulse heights than do neutron initiated events.

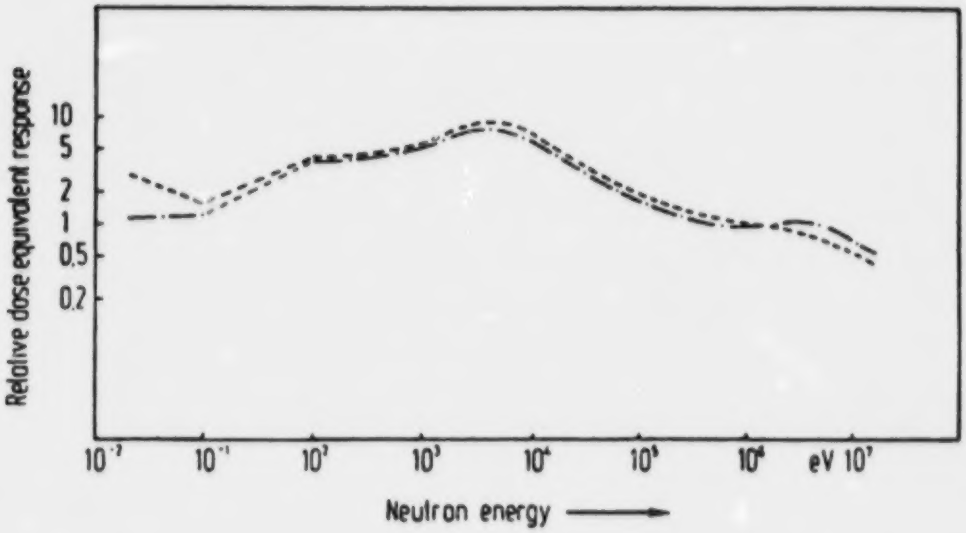
Nearly mono-energetic neutrons from a van de Graaff accelerator were used to assess the accuracy of the TEPC in the energy range 100 keV to 7.5 MeV, (Brackenbush, 1985). On the average, the TEPC measurements agreed with the calculated kerma within 3% with a standard deviation of 9% (Figure 3.2-7). The neutron energy response of the tissue equivalent proportional counter is shown schematically in Figure 5.1-1.

Also, Brackenbush (1985) demonstrated that the quality factor of the neutron radiation can be determined using additional data in the TEPC output spectrum. In the neutron energy range 100 keV to 7.5 MeV, the quality factors estimated using the TEPC compared favourably with the values recommended in ICRU-20 (Figure 3.2-8).

In summary, Brackenbush and others have demonstrated that the TEPC can be used to measure absorbed dose simultaneously for photons and neutrons. Also, it seems possible to calculate neutron quality factor and dose equivalent accurately in the energy range 100 keV to 7.5 MeV. However, the instrument underestimates dose equivalent by 40% at 200 keV and 5 MeV. Overlap of photon and neutron events introduces significant errors only for neutrons with energy less than 10 keV. Some uncertainty persists concerning the comparison of quality factors calculated using TEPC results and those calculated by Monte Carlo computer codes.

Figure 3.2-1

RATIO OF DOSE EQUIVALENT RESPONSE OF A SPHERE OF 30.5 cm DIAMETER RELATIVE TO THAT RECOMMENDED BY ICRP



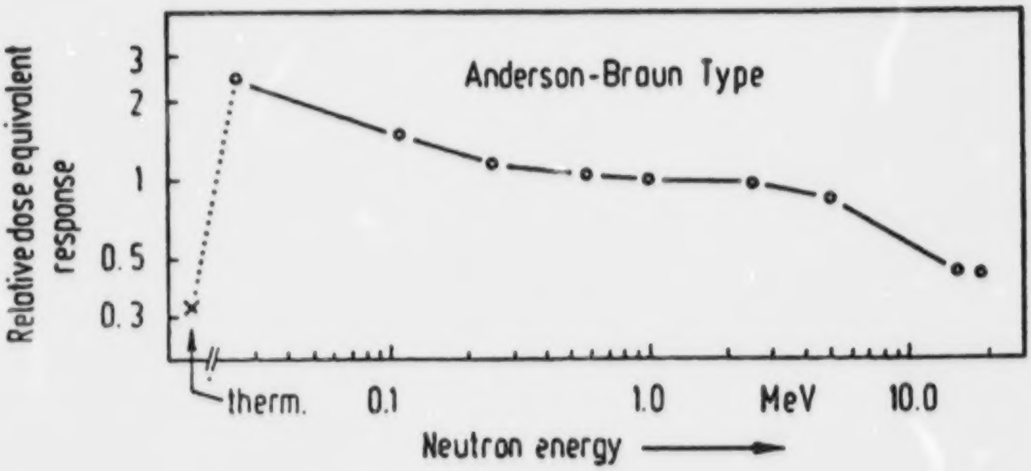
LEGEND:

- - - - EXPERIMENTAL DATA
— - — MONTE CARLO CALCULATION

SOURCE: Cosack, 1985

Figure 3.2-2

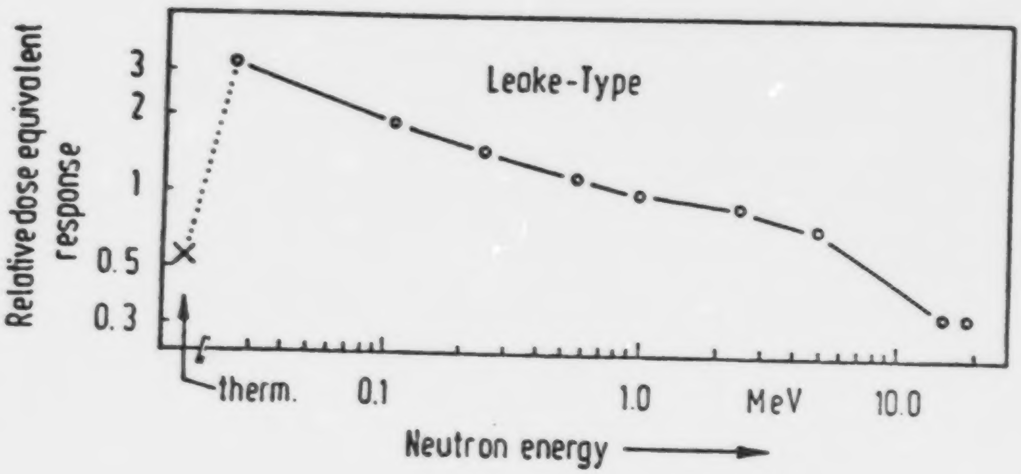
DOSE EQUIVALENT RESPONSE OF THE ANDERSON-BRAUN REM METER RELATIVE TO THE ICRP RECOMMENDATIONS



SOURCE: Cosack, 1985

Figure 3.2-3

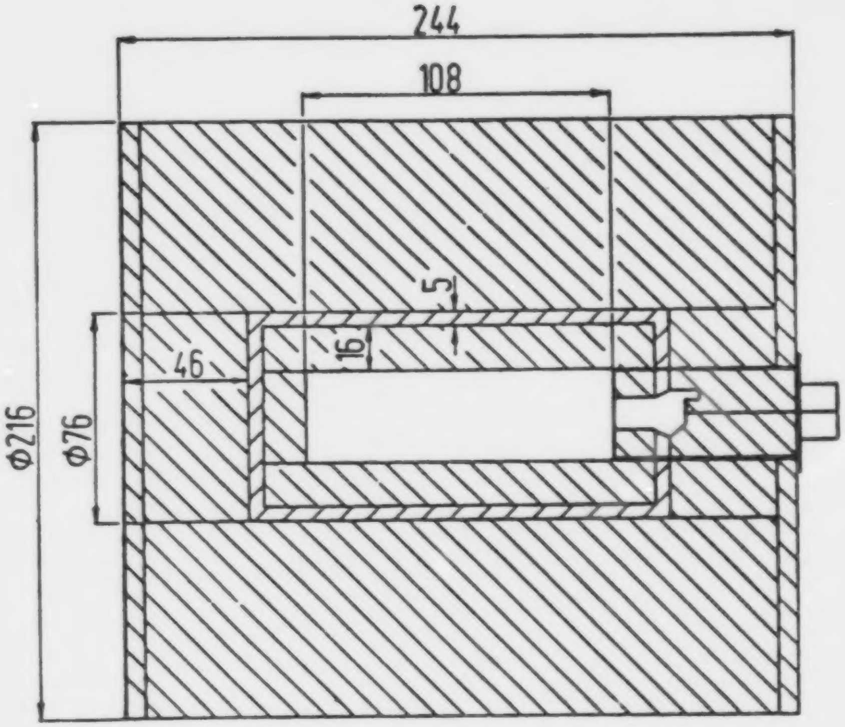
DOSE EQUIVALENT RESPONSE OF THE LEAKE REM METER RELATIVE TO THE ICRP RECOMMENDATIONS



SOURCE: Cosack, 1985

Figure 3.2-4

CROSS SECTION OF ANDERSON-BRAUN TYPE DOSE EQUIVALENT SURVEY METER



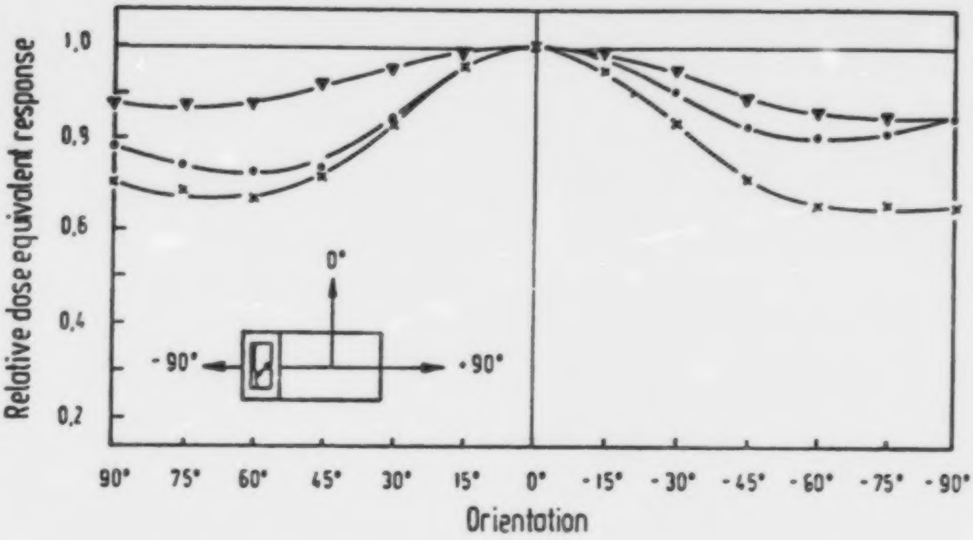
Polyethylene

BF_3-counter
diameter 30×0,5mm
sensitive length 60mm
filling BF_3(94% B^{10}) 600mmHg

Boron plastic
holes Φ10mm over
22% of the area

SOURCE: Cosack, 1985

RELATIVE DOSE EQUIVALENT RESPONSE OF AN ANDERSON-BRAUN SURVEY METER AS A FUNCTION OF ORIENTATION



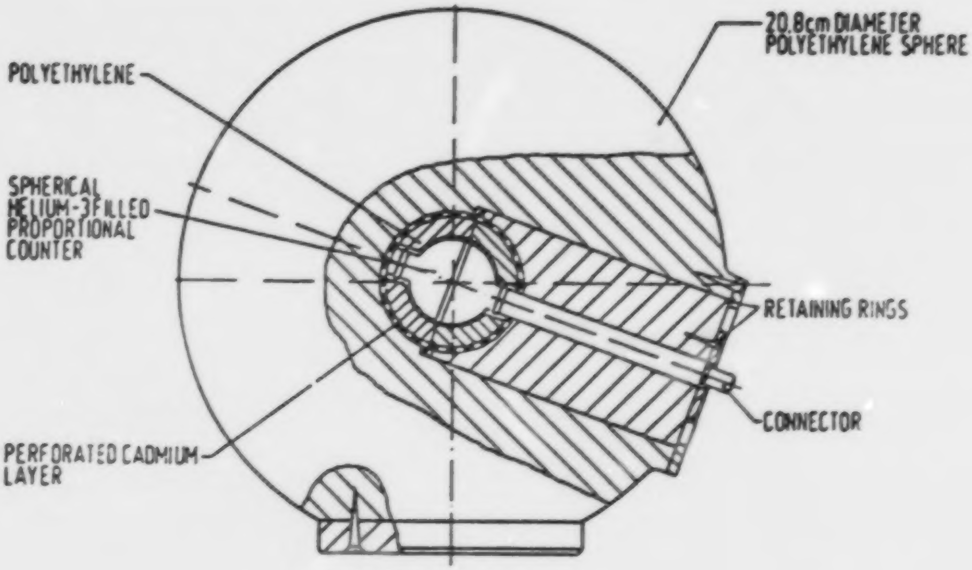
NOTE:.

1. Normalized to 1 at 0° incidence of the neutrons (●) 110 keV, (×) 1 MeV, (▼) 15.5 MeV

SOURCE: Cosack,1985

Figure 3.2-6

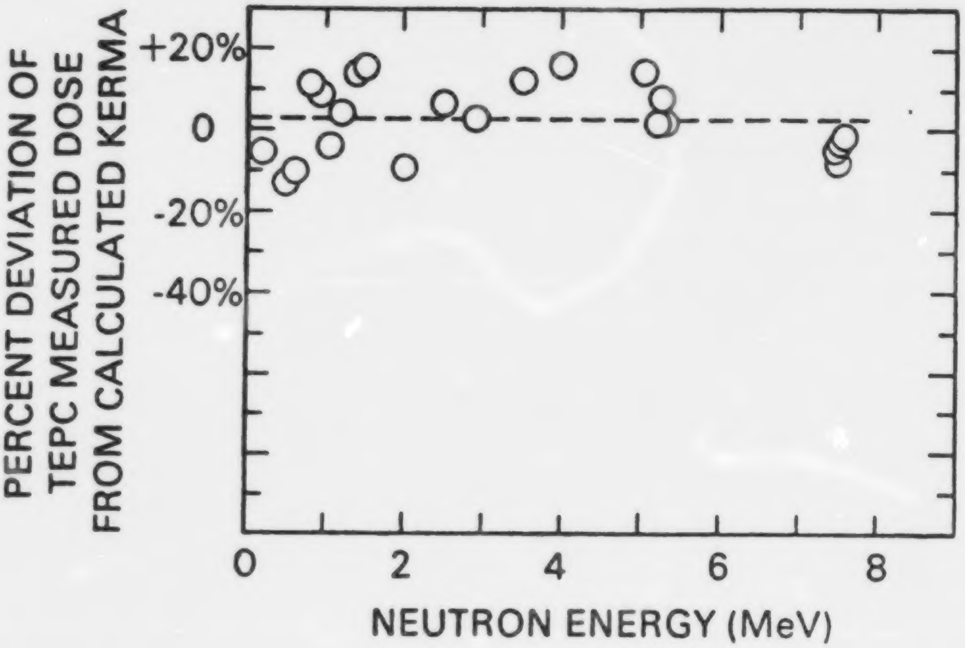
CROSS SECTION OF LEAKE TYPE DOSE EQUIVALENT SURVEY METER


POLYETHYLENE
20.8cm DIAMETER POLYETHYLENE SPHERE
SPHERICAL HELIUM-3 FILLED PROPORTIONAL COUNTER
RETAINING RINGS
CONNECTOR
PERFORATED CADMIUM LAYER

SOURCE: Cosack, 1985

Figure 3.2-7

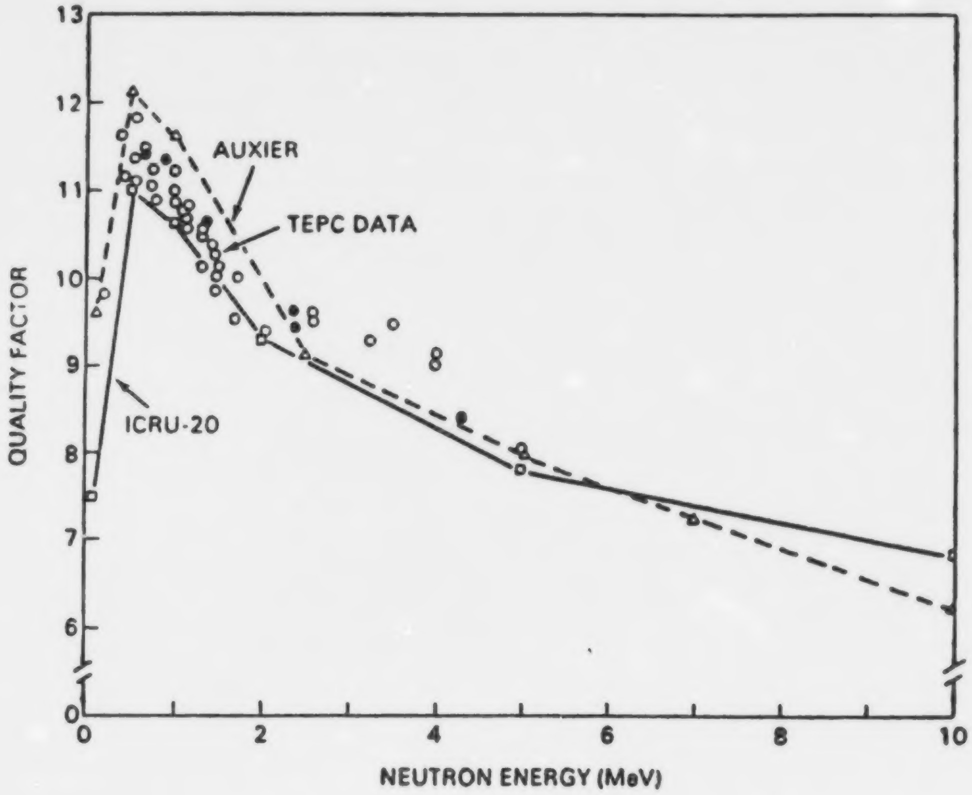
COMPARISON OF NEUTRON DOSE MEASURED BY TISSUE EQUIVALENT PROPORTIONAL COUNTER TO KERMA CALCULATED FROM NEUTRON FLUENCE FOR MONOENERGETIC NEUTRONS



SOURCE: Brackenbush, 1985

Figure 3.2-8

QUALITY FACTOR CALCULATED FROM TISSUE EQUIVALENT PROPORTIONAL COUNTER OUTPUT



SOURCE: Brackenbush, 1985

3.3 Bonner Sphere Spectrometer

One of the most famous and successful neutron flux detectors that was designed for maximum response at specific neutron energies is the Bonner sphere spectrometer. In its original form it consisted of a 4 mm x 4 mm ^{6}LiI (Eu) scintillation crystal which could be positioned at the centre of five polyethylene spherical moderators 2 to 12 inches in diameter. The form has now been adapted to include a central detector that may be a BF_3 or a ^{3}He detector or a set of activation foils. Also, the number of spectrometer spheres now used is usually six.

When a neutron detector is placed at the centre of a set of spherical moderators of various diameters each detector-moderator combination will have a different response to neutrons as a function of energy. The response for a set of moderator spheres with a 4 mm x 4 mm ^{6}LiI (Eu) scintillator is shown in Figure 3.3-1 (Awschslom, 1985).

The response functions for several of the different geometry/detector combinations for spectrometers have been calculated using Monte Carlo computer codes. These calculations and other factors contribute to errors and uncertainties in unfolding the data to calculate the original neutron spectrum. For instance, the density of commercial polyethylene ranges between 0.91 and 0.96 g/cm^3. The response ratio for 10 inch diameter polyethylene spheres having densities 0.90 and 0.95 g/cm^3 is shown as curve I in Figure 3.3-2. The response ratio varies by approximately 40% while the densities differ by only about 6% over the same range of neutron energies. Another source of error arises when polyethylene responses are used for different moderating material. Curve II in Figure 3.3-2 represents the response ratio for 10 inch diameter polyethylene and water spheres. At neutron energies below about 0.1 MeV the response ratio is about a factor of 3. Curve III in Figure 3.3-2 represents the response ratio for a 2 inch diameter polyethylene sphere surrounding a ^{6}LiI(Eu) scintillator with dimensions 4 mm x 4 mm or 12.7 mm x 12.7 mm. Over the neutron energy range from thermal to 100 MeV, the response ratio varies from 16% to 4%. Because of the difficulty, time and cost involved in calculating Bonner sphere response functions, the results of some experimental determinations of neutron spectra are calculated using Bonner sphere response functions based upon compositions, densities or central cavity dimensions that only approximate the characteristics of the spheres used in practice. Consequently, the uncertainties associated with the calculated neutron spectra can be large as demonstrated above.

The measurement of the neutron spectrum at just one location requires a total of seven or more separate measurements at the location depending upon the number of spheres used. Attempting a large area survey with such an extensive set of detectors is obviously a large task.

Neutron detectors used in the central cavity of the Bonner spheres may be either active or passive. In either case, the gamma sensitivity of the detector must be investigated and compensated by suitable means if the interference with the neutron response is unacceptable. Active detectors may be either scintillator/photomultiplier combinations or gas filled counters operated in the proportional or Geiger modes. Their main advantage is real time data acquisition and in some cases good gamma discrimination.

Passive detectors used in Bonner sphere spectrometers may be based on either radioactivation or radioluminescence principles. They have several advantages over active detectors in that they are insensitive to temperature, humidity, and RF interference and they can integrate over the duty cycle of an accelerator.

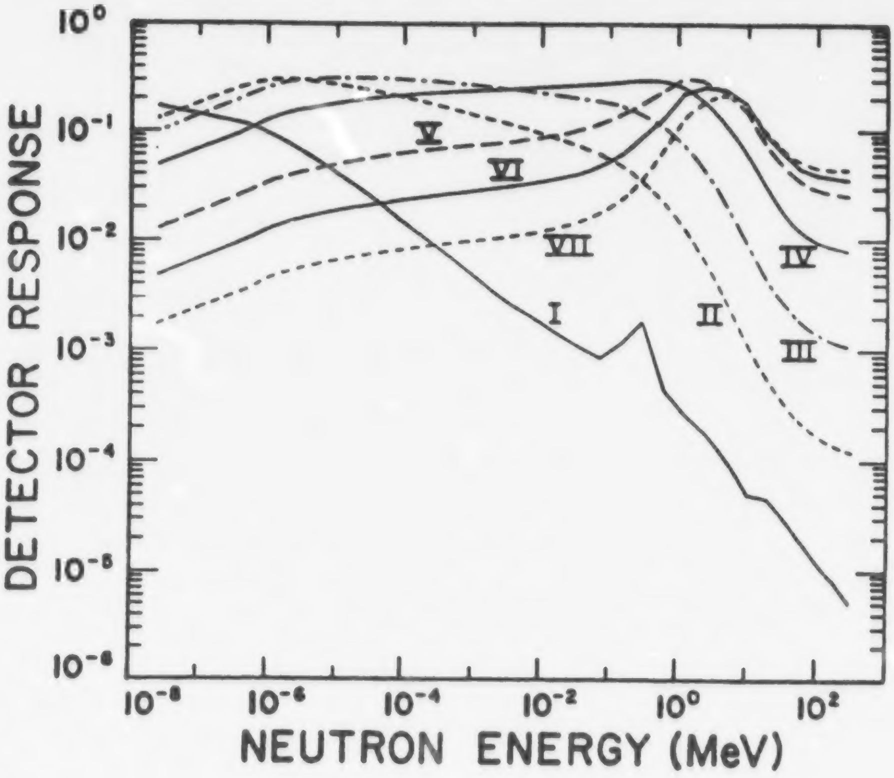
Spectral deconvolution is the analytical exercise involved in estimating the original neutron energy spectrum from the results of the multisphere measurements. This task introduces errors and uncertainty into the result in addition to those already identified. The response functions are usually based upon Monte Carlo or similar computer models and are imperfect estimates of the true function.

Calibration of the Bonner sphere spectrometer is usually carried out using Am241-Be, Am241-B or Cf252 sources. To calibrate with such a source one must know the total neutron output, source anisotropy or self shielding and the neutron energy spectrum of the source.

The major advantage of the Bonner sphere spectrometer is that it covers the energy range from thermal to tens of MeV, a range that no other method can duplicate. Unfortunately, Bonner spheres have inherently low energy resolution and are awkward to position because of their size and weight. Furthermore, if only one detector is used, each moderating sphere must be irradiated sequentially and the stability of the neutron source may be of concern throughout the duration of the measurements. The time loss between exposures may be appreciably depending upon the time required a)to transfer the neutron detector from sphere to sphere and b) to relocate the sphere for irradiation. Diversion of the accelerator beam or reduction in beam current is a prudent practice during the placement of the Bonner sphere in the irradiation position. Alternatively, if several detectors are used coincidentally, the intercalibration among the detectors and the mutual interference among the spheres are potential sources of uncertainty and error.

Figure 3.3-1

NEUTRON FLUENCE RESPONSE OF A MULTISPHERE SPECTROMETER WITH SPHERE DIAMETERS FROM 2 TO 12 INCHES



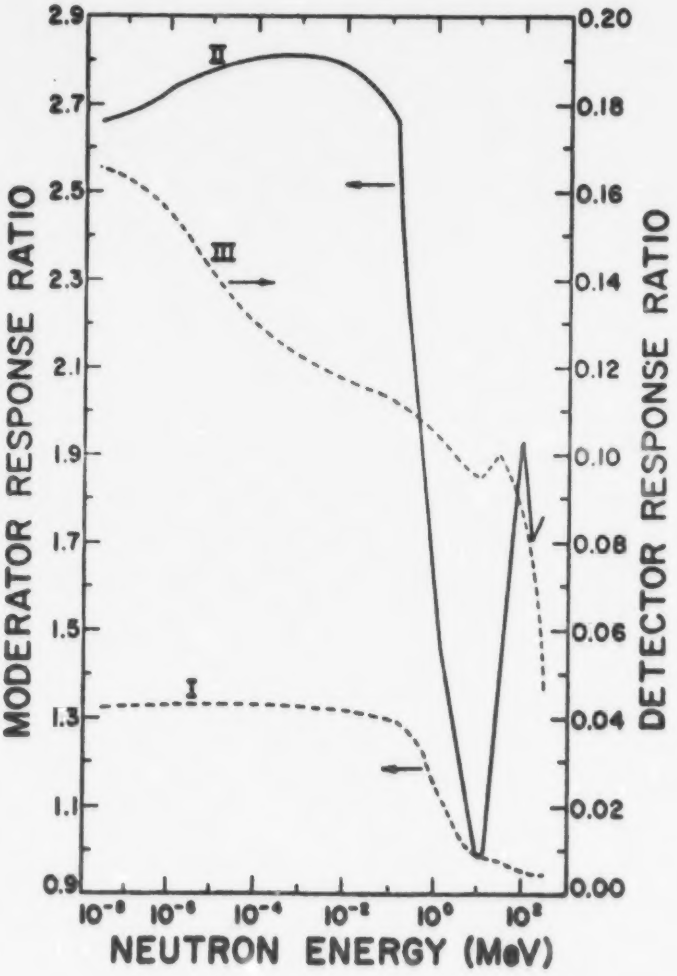
MODERATOR DIAMETERS:

I = 0
II = 2 inches
III = 3 inches
IV = 5 inches
V = 8 inches
VI = 10 inches
VII = 12 inches

SOURCE: Awschalom, 1985

Figure 3.3-2

COMPARISON OF RESPONSES FOR TWO DIFFERENT MODERATOR DENSITIES



NOTE:

1. I = Response ratio for two densities (S = 0.90 and 0.95 g/cm^3) of 10" diameter polyethylene spheres
2. II = Response ratio polyethylene and water spheres, 10" diameter
3. III = Response ratio for 2" diameter polyethylene sphere containing 4 mm x 4 mm or 12.7 mm x 12.7 mm ^{6}LiI scintillation detector

SOURCE: Awschalom, 1985

3.4 ""Delta" Spectrometer

The "Delta" spectrometer, developed by the Central Electricity Generating Board at Berkeley Nuclear Laboratories, England was adopted by Central Health Physics Services for use by Ontario Hydro. Briefly, the instrument consists of a 30 cm long moderating cylinder of polyethylene shielded against neutrons on all surfaces except the front end. In a removable core, ^{6}LiF TLDs are used to measure the linear distribution of thermal neutrons along the cylinder axis. Simultaneously, the gamma contribution is determined by the measurement of ^{7}LiF TLDs. The thermal neutron distribution along the axis is determined by the spectrum of neutrons incident on the front end. A computer program uses the response from previous monoenergetic neutron calibrations to unfold the measured response to obtain the unknown neutron spectrum.

The neutron energy response of the "delta" spectrometer is shown schematically in Figure 5.1-1.

4.0 PERSONAL NEUTRON DOSIMETERS

Track Etch Detectors

A neutron dosimeter suitable for both personal and area monitoring that is gaining wide acceptance is the track etch detector. Neutron interactions are registered in an insulating medium by charged particles recoiling from neutron collision. The damaged insulating medium is etched by a caustic solution to yield tracks which are visible under low magnification. The track density is multiplied by an appropriate calibration factor to yield neutron dose equivalent.

Most tracks produced by neutrons in track etch detectors are from recoil protons, originating either in the surface layer that is removed by etching or in a hydrogenous radiator placed against the front of the track etch detector. Certain plastics (polycarbonate, CN-85 (cellulose nitrate, manufactured by Kodak-Pathé, France) and CR-39 (allyl diglycol carbonate, manufactured by Pershore Moulding Ltd., UK, American Acryllics Corp., USA, MOM-Atomki, Hungary, Track Analysis Systems Ltd., UK)) have been identified as having the characteristics of good track registration. CR-39 track etch detectors are currently the most promising personal neutron dosimeters and have sensitivity to a wider range of neutron energies than the others. Therefore, the current state of development of CR-39 is described below.

One of the main goals of neutron dosimetry has been to develop a dosimeter with an energy independent response that extends below the low energy cut-off of about 0.5 MeV for nuclear emulsions and fission fragment damage track detectors. Early use of CR-39 reduced the lower neutron energy limit to 100 to 200 keV but the response was too energy dependent (Cross, 1987). In 1983, Tommasino introduced high temperature electrochemical etching and the dose equivalent energy response was greatly improved.

The dose equivalent energy response has been measured by many researchers during the development of CR-39 over the past five years (Cross, 1987) and is shown in Figure 4.1-1. These were all obtained using electrochemical etching but with a variety of techniques and using CR-39 from different manufacturers. There is a wide variation in response with some investigators achieving a relatively flat response over the energy range 50 keV to almost 10 MeV.

Electrochemical etching as replaced chemical etching as a means of reducing the lower limit of sensitivity to neutron energy. The chemical process etches the damage tracks in preference to the bulk of the foil and obliterates tracks made by low energy protons (recoils from low energy neutrons). Whereas, the electrochemical etching process enlarges all damage tracks caused by protons corresponding to the energy range 100 keV to 3 MeV.

The effect on dose equivalent response as a function of incident neutron angle has been reported for four neutron energies by Matiullah, 1987a (Figure 4.1-2). The dose equivalent response is strongly influenced by incident neutron angle and decreases by 94% when the angle is changed from perpendicular to parallel to the CR-39 surface. In this experiment a composite radiator (150 um polyimide, 800 um polystyrene and 5000 um polyethylene) was fixed to the front surface of the CR-39 detector. However, the composite radiator did not improve the response to neutron angle over the response of a bare CR-39 or single radiator/CR-39 combination. The composite radiator was an optimum design to yield the flattest energy response for neutrons incident normally (James, 1987). Using the radiator, the variation in the dose equivalent response over the energy interval 0.1 to 19 MeV is reduced to less than a factor of 2 (compared to a factor of 12 for the bare detector).

One solution to the problem of the directional dependence of the dosimeter was the construction of a detector assembly consisting of 3 CR-39 detectors, each with its radiator stack, placed at right angles to each other. The average dose equivalent of the three detectors varies by only ±23% over the range of incident neutron angles from 90° to 0° (Matiullah, 1987a).

Hankins (1987) illustrated a simple method to improve the angular response of the CR-39. When it is covered on both sides by polyethylene radiators, the ratio of responses at 90° and 0° incidence is improved by more than 40%. This is accomplished by measuring tracks on the rear surface of the CR-39, rather than on the front.

The CR-39 detector was modified by Afarideh et al. (1987) to extend the energy response to include thermal neutrons by placing the detector in intact with $Li_2B_4O_7$ or boron converter screens. The presence of thermal neutrons was detected by etching the alpha tracks in the detector caused by the neutron alpha reactions on ^6Li and boron in the convertors. Afarideh et al. (1987), demonstrated that the number of tracks was linearly proportional to the total thermal neutron fluence over the range 10^8 to 3×10^9 n cm^{-2}.

Greenhouse (1987) reported the results of a comparison of CR-39 with NTA film in an accelerator environment. A beam of argon ions at 8.5 MeV per nucleon was targeted on a beam stop at about 6 metres distant from a phantom to which the dosimeters were attached. The dose equivalent measured by an Andersson-Braun rem meter, NTA film dosimeters and the CR-39 detectors were compared to the dose equivalent measured with the Bonner sphere system. The dose estimated using the CR-39 detector was only 12% less than the Bonner sphere result. Whereas, the doses estimated using the Andersson-Braun meter and the NTA films were 47% larger and 25% smaller than the results with the Bonner spheres, respectively.

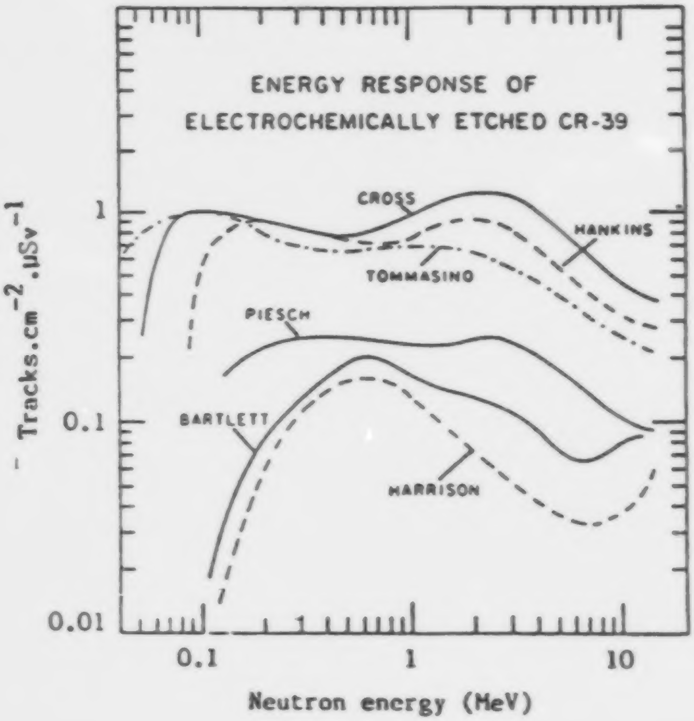
Greenhouse identified other potential advantages of the CR-39 over the NTA film including lack of latent track fading, the potential for automated track readout and improved response to lower energy neutrons. Bartlett (1987) has reviewed the ageing of and environmental effects on CR-39 detectors. Ageing, the change in detector properties during storage at ambient conditions between manufacture and irradiation, has been observed by several researchers. Generally, it has been found that for storage in air at room temperature, background increases with time and sensitivity (both to neutrons and alpha particles) decreases. The magnitude of these effects depends on the plastic type and processing method. Ageing effects can be avoided by "pre-aging" plastic or by applying an average correction factor which is estimated to be less than 5-10% per month (Bartlett, 1987)

Alpha tracks from environmental radon and radon daughters are not a problem as they can be simply eliminated by pre-etching or by keeping the plastic covered.

No neutron data is available for the effect of temperature on CR-39 during irradiation (Bartlett, 1987). However, indications from etching of alpha tracks are that for the range of temperatures, from 10 to 40 degrees C, the variation in observed response is likely to be less that +/- 15%. At moderate temperatures, 30 degrees C or less, long term fading does not present a problem. However, more information is required concerning the effects of temperature on the first few weeks post irradiation and at higher temperatures in the longer term.

Figure 4.1-1

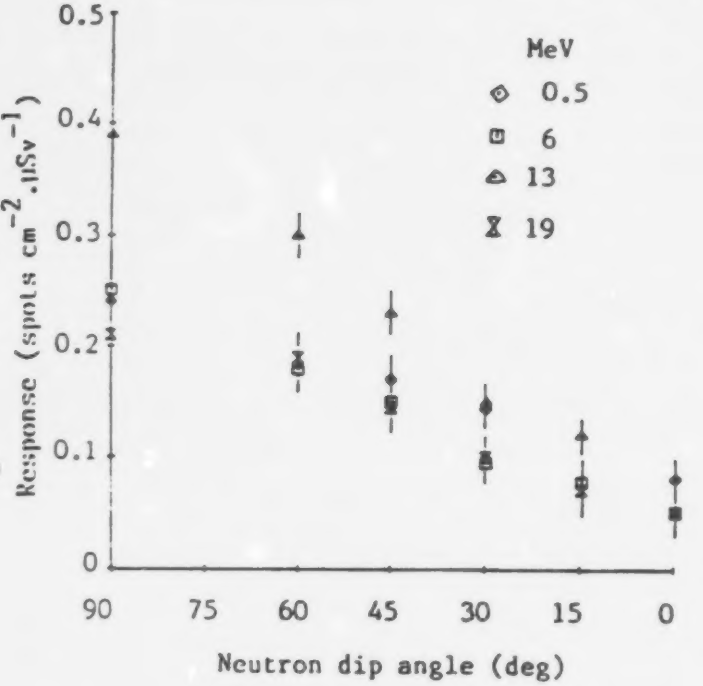
THE DOSE EQUIVALENT RESPONSE OF CR-39 (USED WITH THE FRONT RADIATOR STACK) AS A FUNCTION OF NEUTRON ENERGY



SOURCE: Cross, 1987

Figure 4.1-2

THE DOSE EQUIVALENT RESPONSE OF CR-39 AS A FUNCTION OF INCIDENT NEUTRON ANGLE



SOURCE: Matiullah, 1987a

4.2 Superheated Liquid Detectors

4.2.1 Neutron Bubble Detector

A different approach to neutron dosimetry using superheated liquid droplets suspended in a polymer matrix is under development by Ing (1986). Very small superheated liquid droplets are uniformly dispersed in a firm polymer which is contained in a small capped and sealed glass or plastic test tube. Under neutron irradiation, the droplets vapourize to form small visible bubbles, which are immolized in the polymer. The number of bubbles is proportional to the neutron dose equivalent and they are counted by the naked eye or under low magnification. These detectors can be designed to be extremely sensitive, detecting down to about 1 uSv and can be varied in detector sensitivity and neutron energy response over orders of magnitude.

The detectors are small, reusable, require no external power source and are suitable for use as a personal dosimeter as well as an integrating area monitor. As personal dosimeters they have the added feature of providing an immediate visible indication of the neutron dose equivalent and have an inherently isotropic response.

The neutron energy response is reported to be ± 20% over the range 200 keV to 14 MeV (Figure 4.2-1). However, Perks (1987) and Ipe (1987) report a much wider variation over the same energy range (Figures 4.2-2 and 4.2-3, respectively). Over the energy range 20 keV to 3 MeV Perks (1987) reports that the dose equivalent response varies by a factor of about 8. Ipe (1987) measured the dose equivalent energy response over the energy range 0.4 to 4.5 MeV and reported a variation of about a factor of 3. Detector response to thermal neutrons is about 20% of the response in the neutron energy range noted above (Ing, 1986).

Detector response has not been measured to neutron energies in the range thermal to 20 keV.

The detector background (i.e. number of bubbles) at initial sensitization is zero under normal circumstances.

The detector is reported to be insensitive to gamma radiation, with the exception of gamma rays of sufficient energy to generate photoneutrons. This was verified by Perks (1987) who reported that no bubbles were observed in detectors irradiated to 22 mSv using ^{137}Cs gamma radiation. Also, Ipe (1987) demonstrated that the BD-100 bubble detector (Bubble Technology Industries) was insensitive to gamma radiation from ^{60}Co for doses up to 15 Gy. However, he observed the production of bubbles (likely from photoneutrons) when the detector was placed in a 15 MeV electron beam.

The stability of the neutron bubble detectors system has been demonstrated by Ing (1986) (Figure 4.2-4). Unsensitized detectors can be stored for a year before use, with no loss in sensitivity. Ipe (1987) has corroborated this for storage times up to 5 months. Also, Ipe (1987) noted that there was no significant fading of bubbles up to 5 months after irradiation. Ing (1986) claims that the useful life of the detector is four weeks in the sensitized state. However, Ipe (1987) reports that the polymer in the detector deteriorates after 2 weeks in the sensitized state.

The neutron bubble detectors have been shown not to be impact resistant (Ipe, 1987). When dropped on the floor after irradiation, clusters of tiny bubbles were produced in the detector. Some dose information may still be retrievable as larger bubbles are produced by neutron irradiation than by physical shock.

The detectors are unaffected by humidity, but their response is sensitive to temperature. The sensitivity of the detector depends upon the vapour pressure of the droplets which in turn depends upon the temperature. In early models of the detector, dose equivalent sensitivity to temperature increased by about a factor of four from 10° C to 40° C in a non-linear fashion (Ing, 1989) (uncompensated curve in Figure 4.2-6). Temperature correction of the data is further complicated by the temporal lag experienced by the detector in equilibrating with its surrounding temperature. Researchers at Bubble Technology Industries recently reported that they have successfully reduced the dose equivalent sensitivity to temperature to an increase of only about 45% in the range 10° to 40°C (Ing, 1989).

Based upon the bubble detector design, Bubble Technology Industries has developed a neutron spectrometer for use over the energy range 30 keV to 14 MeV. The proprietary composition of the bubble detector can be altered to produce detectors with 6 different neutron energy cut-offs (Figure 4.2-5). These detectors offer a simple method for characterizing the neutron energy spectrum.

In conclusion, the neutron bubble detector has several properties, such as high sensitivity, low threshold of detection and isotropic response, that make it a desirable neutron detector. However, some characteristics that influence its response, such as sensitivity to impact and temperature variations may limit its use as a practical neutron dosimeter in some cases.

4.2.2 Superheated Drop Detector

The superheated drop neutron detector (SDD) (Apfel, 1985, 1989) is based upon a principle similar to that of the bubble chamber.

A suitable low boiling point fluid is dispersed in drop form throughout a viscous gel or polymer which holds the fluid drops in superheated form. When charged particles from neutron interactions deposit sufficient energy in the drops, they vapourize and float to the surface of the gel. The volume of vapour or the acoustic waves given off when superheated drops vapourize serve as a measure of neutron field intensity. Superheated drops can be maintained for several months in the semi-solid gel matrix. Repressurization needed in the conventional bubble chamber is avoided and when one drop boils explosively, it will not normally trigger boiling in adjacent drops. The SDD can operate at room temperature, is portable and requires no external power source.

Four kinds of superheated fluids have been tested by Apfel (1989), Freon-12*, Freon-114, Freon-142B and Isobutane. A piezoelectric transducer was fixed to each of the detectors and acoustic signals were analyzed electronically when a superheated drop vapourized. Most of the energy of the acoustic signal is in the range of 1-10 kHz and persists for 20-30 ms.

* Freon is a registered trademark of the Dupont Co.

Experimental results of energy response measurements at room temperature (20-23°C) demonstrate that Freon-12 is sensitive to neutrons in the energy range 0.025 MeV to 14 MeV. The energy response trend has some similarity to the ICRP dose equivalent curve shown in Figure 4.2-7. The detector response is within 40% for neutrons above 100 keV and within a factor of 10 for energies below 100 keV (Apfel, 1989).

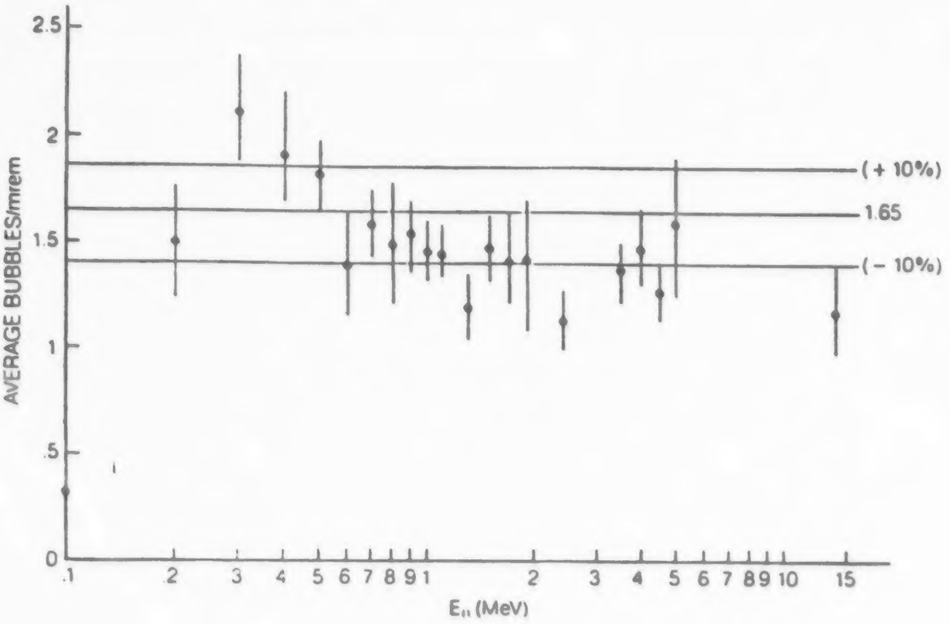
Other materials tests by Apfel (1989) showed quite different responses. Isobutane has a neutron energy threshold at about 1.2 MeV, Freon-142B at about 2.1 MeV and Freon-114 at about 6 MeV. All three were less sensitive to neutrons than Freon-12 but exhibited similar neutron response up to the maximum test energy of 14 MeV.

The superheated drop detector has several positive features. The cylindrical symmetry of the superheated drop detector results in 100% isotropic response to neutrons. Neutron produced bubbles are visible to the naked eye, giving direct reading capability. The detector is insensitive to gamma radiation, as long as the energy is less than about 6 MeV.

Unfortunately, the superheated drop detector has some negative features. When operated in the acoustic pulse counting mode the pulse duration (and dead time) is 20 to 30 ms. This restricts the maximum dose rate to much less than 9 mSv/h, in order to avoid pulse pile-up. Finally, the dose equivalent response is strongly dependent upon temperature and varies at the rate of 5.3%/^{o}C.

Figure 4.2-1

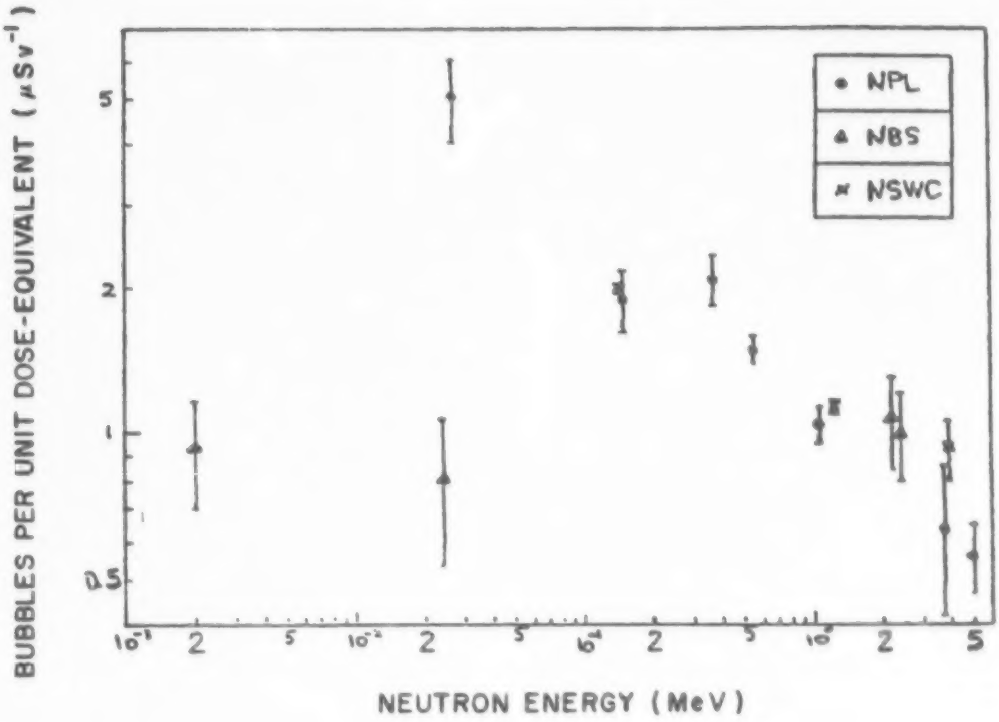
MEASURED DOSE-EQUIVALENT ENERGY RESPONSE OF BUBBLE DETECTORS



SOURCE: Ing, 1986

Figure 4.2-2

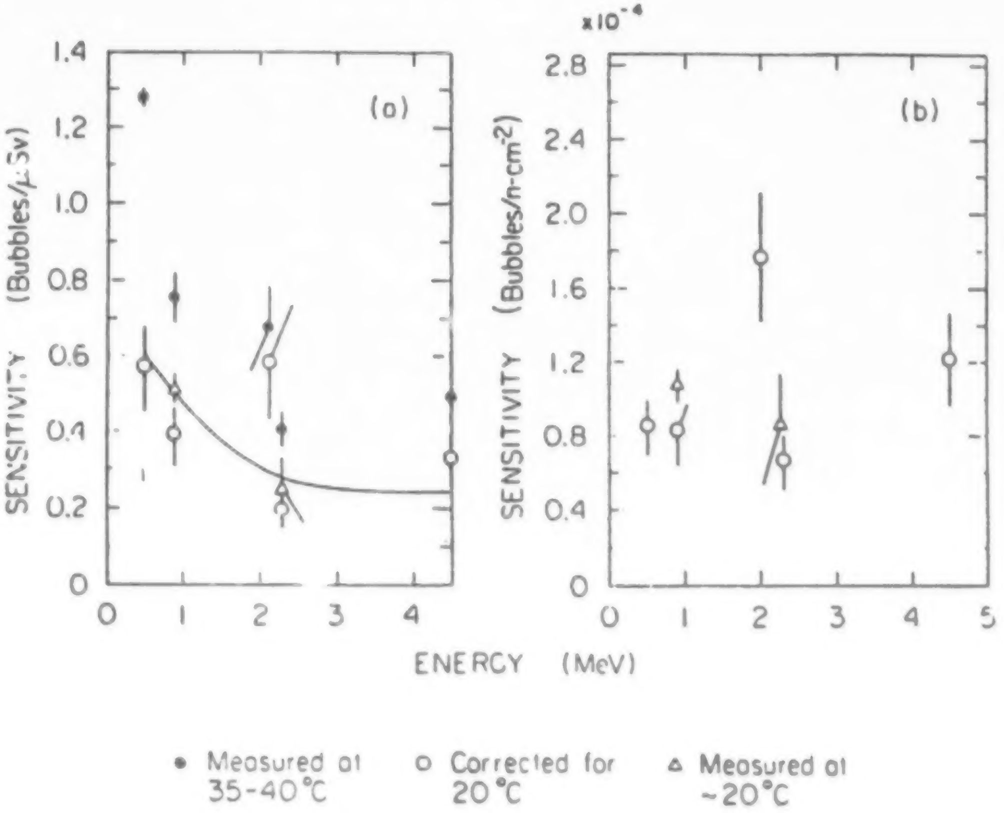
MEASURED DOSE-EQUIVALENT ENERGY RESPONSE OF BUBBLE DETECTORS



SOURCE: Perks,1987

Figure 4.2-3

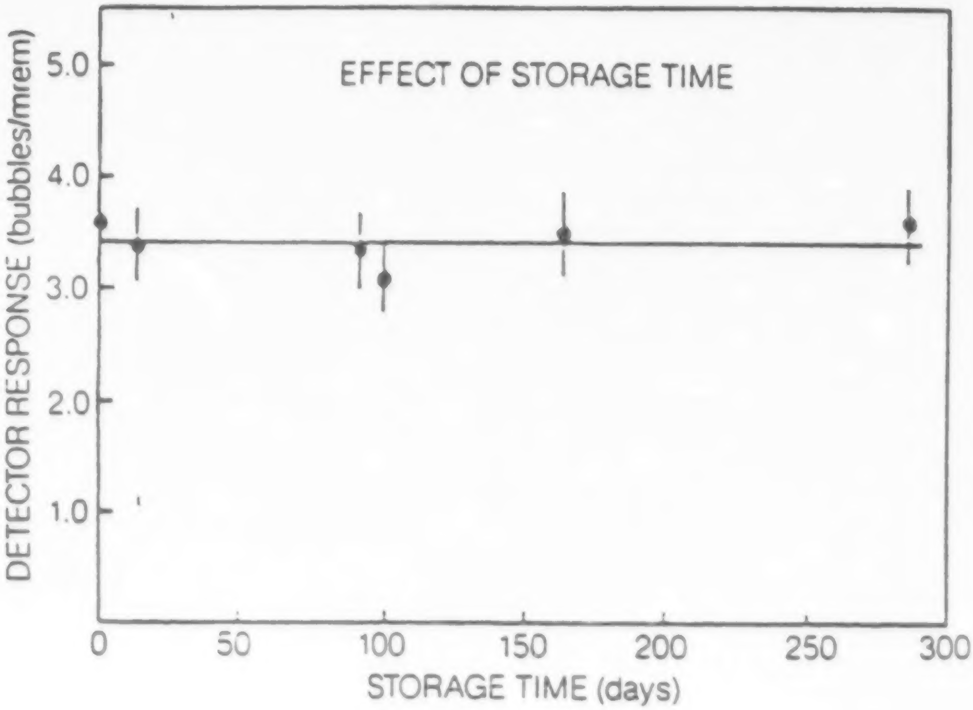
MEASURED DOSE-EQUIVALENT AND FLUENCE ENERGY RESPONSE OF BUBBLE DETECTORS



SOURCE: Ipe, 1987

Figure 4.2-4

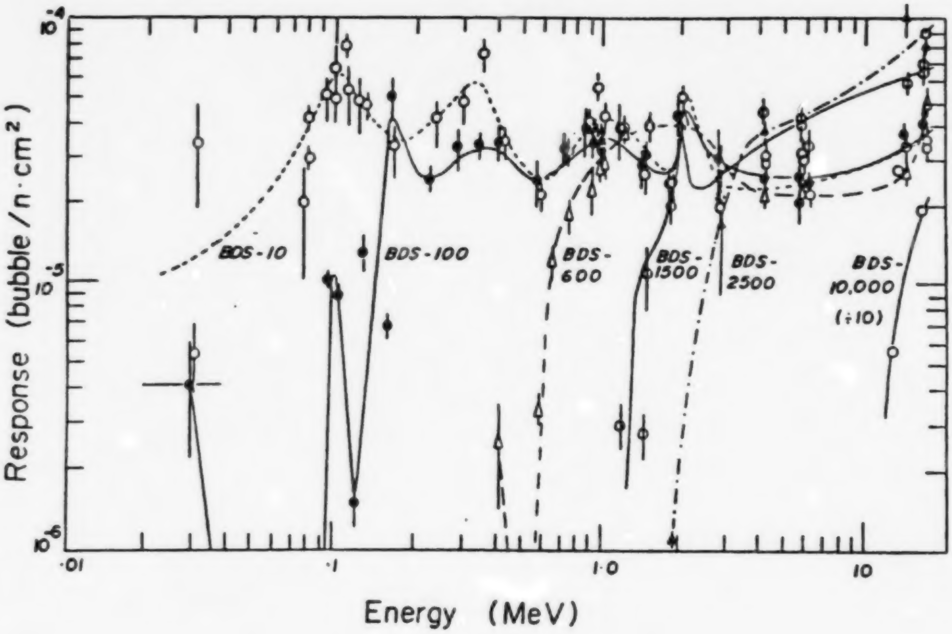
STABILITY (SHELF LIFE) OF UNSENSITIZED BUBBLE DETECTORS



SOURCE: Ing,1986

Figure 4.2-5

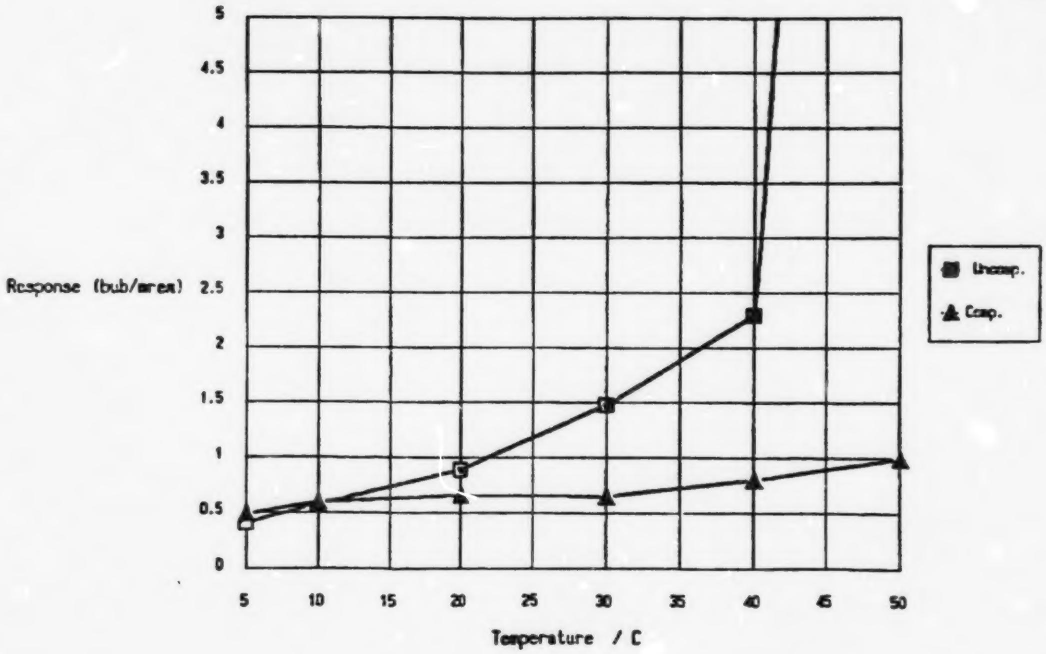
BUBBLE DETECTOR SPECTROMETER



SOURCE: Bubble Technology Industries, 1988

Figure 4.2-6

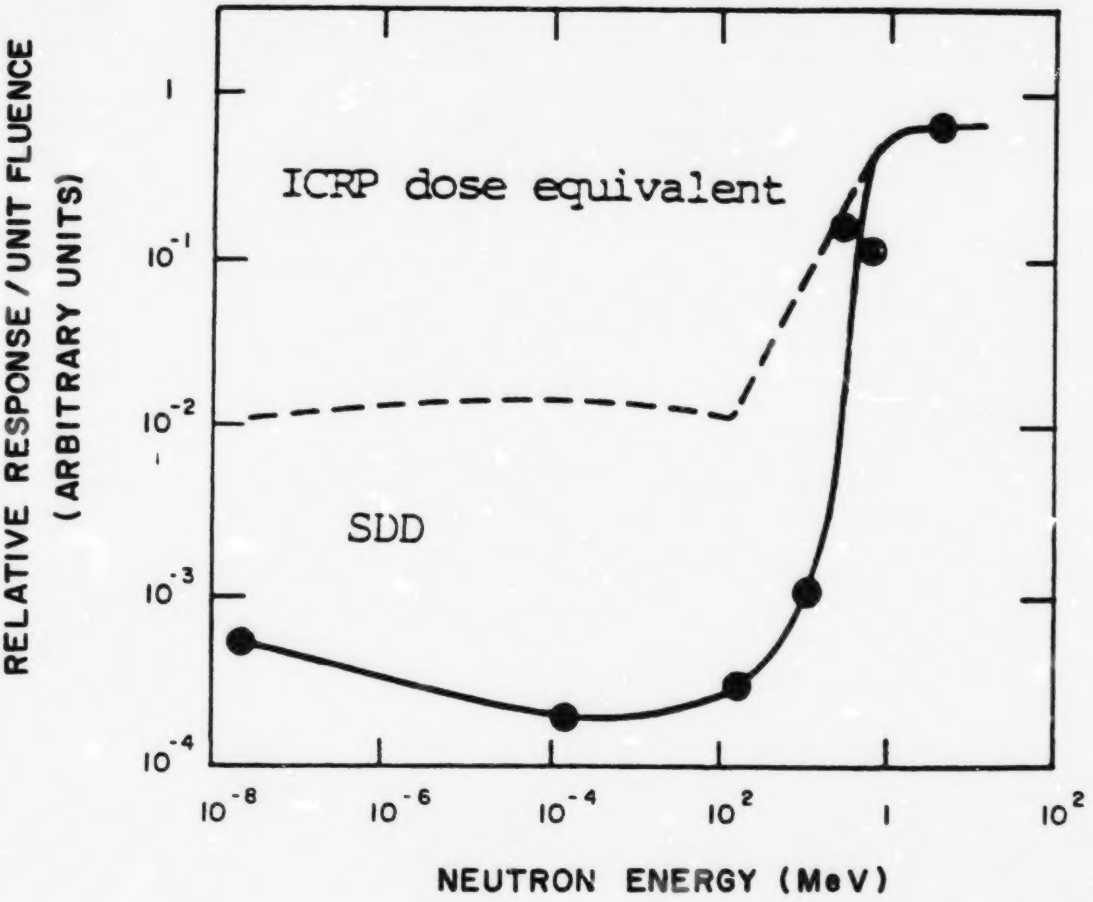
MEASURED DOSE EQUIVALENT TEMPERATURE RESPONSE OF BUBBLE DETECTOR



SOURCE: Ing, 1988

Figure 4.2-7

COMPARISON OF THE NEUTRON FLUENCE ENERGY RESPONSE
OF AN IDEAL DOSIMETER
WITH THE PRESENT SUPERHEATED DROP DETECTOR



SOURCE: Apfel, 1985

4.3 Albedo Neutron Dosimeters

Neutrons incident upon the body will be moderated and scattered as a result of interactions within the body. The albedo neutron dosimeter detects the low energy neutrons that are reflected from the body, usually by means of LiF thermoluminescence dosimeters (TLDs). Two TLD detectors are used, one containing ^{6}Li and the other ^{7}Li. The latter is much less sensitive to neutrons and is therefore, assumed to respond only to gamma rays. The TLD containing ^{6}Li responds to both neutrons and gamma rays. In the ideal case, the difference between the two detectors can be used as a measure of low energy neutrons thermalized in the body and indirectly the incident high energy neutron fluence.

Unfortunately, the efficiency with which the body (or phantom) thermalizes the incident neutrons and scatters them back to the albedo detectors decreases with increasing neutron energy. The response per Sv of neutrons incident on the body is about 100 times greater for thermal neutrons than for neutrons of energy about 1 MeV. This problem of energy dependence has been overcome by calibrating the dosimeters in the working environment. This solution introduces its own drawback when the wearer of the albedo dosimeter moves about to alternate locations where the neutron spectrum is different and therefore, different calibrations apply.

Piesch and Burgkhardt (1985) assessed several albedo dosimeter types. They found by calibration with monoenergetic neutrons that the albedo response of all dosimeter types were similar in energy dependence and differences were mainly for thermal and epithermal neutrons (Figure 4.3-1). The large decrease of the albedo response above 10 keV illustrates the restrictions on applying these dosimeters to personal monitoring when high energy neutrons are present.

Compared with TLDs, albedo track etch detectors with converters of ^{6}Li or ^{10}B layers in contact with Makrofol or CR-39 are gamma insensitive, highly sensitive to neutrons, but show, on the other hand, high random errors, a relatively high angular response and a limited dose range. Consequently, the following discussion focuses on TLD-albedo dosimeters.

The angular response of the Karlsruhe albedo dosimeter is shown in Figure 4.3-2. The angular response is quite severe, dropping to 35% at 90 degrees to the forward direction and less than 10% at 180 degrees. The Universal albedo dosimeter response drops to only 70% at 90 degrees to the forward direction (Piesch and Burgkhardt, 1985). In comparison the track etch angular response is as poor as the Karlsruhe dosimeter.

The lowest detectable neutron dose for the Karlsruhe albedo dosimeter was found to be 0.05 mSv at reactors and 0.02 mSv at DT generators. However, it could measure dose equivalent up to 1 Sv.

For comparison, the recoil track detectors could measure neutron dose in the range 0.1 mSv to 300 mSv and the radiator track detector could measure from 2 uSv up to 10 mSv.

The albedo neutron fluence changes significantly with the distance of the albedo dosimeter from the body. The relative neutron dose equivalent response is shown in Figure 4.3-3.

As noted above, site specific calibrations must be carried out for application of the albedo dosimeter in the field. The contribution of stray and scattered neutron radiation may change the calibration factor by up to an order of magnitude. The calibration can be carried out by placing the albedo dosimeter on a phantom and comparing the result to measurements using a dose equivalent type meter (rem meter). This method is not recommended if intermediate energy neutrons are present as it may introduce an error inherent in the rem meter which overresponds by a factor of 5 to intermediate energy neutrons depending upon the neutron energy spectrum. A more accurate method, using two moderating spheres with diameters of 23 and 7.5 cm and the thermal to epithermal ratio by using a cadmium shield around the small sphere, has been devised by Gibson (1979).

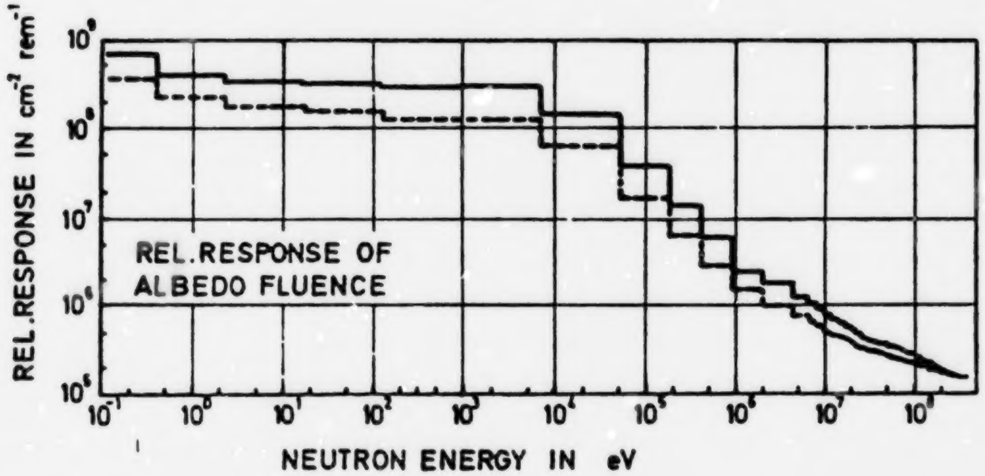
The results of intercomparison experiments among different laboratories are also informative because they audit the readout and quality assurance programs of the laboratory, rather than the dosimetry system itself. The preliminary results (Piesch, 1985) obtained with the Karlsruhe albedo dosimeter in the Euratom Intercomparison Experiment in 1982 showed a relative standard deviation of neutron dose measurement of 14%. This is the magnitude of uncertainty which may generally be expected in albedo neutron dosimetry.

Piesch (1985) reports that the mean energy of the stray neutron fields in experimental and power reactors is about 0.5 MeV. However, the mean value at any location may vary over the range from 0.05 Mev to 1 MeV and the dose contribution from thermal neutrons may vary from 1 to 60% of the total.

Piesch (1985) concludes that experience over the last 10 years indicates that the location-dependent response of albedo dosimeters at a facility can be estimated within a reasonable margin of error of about +/- 30% if additional neutron detectors are used.

Figure 4.3-1

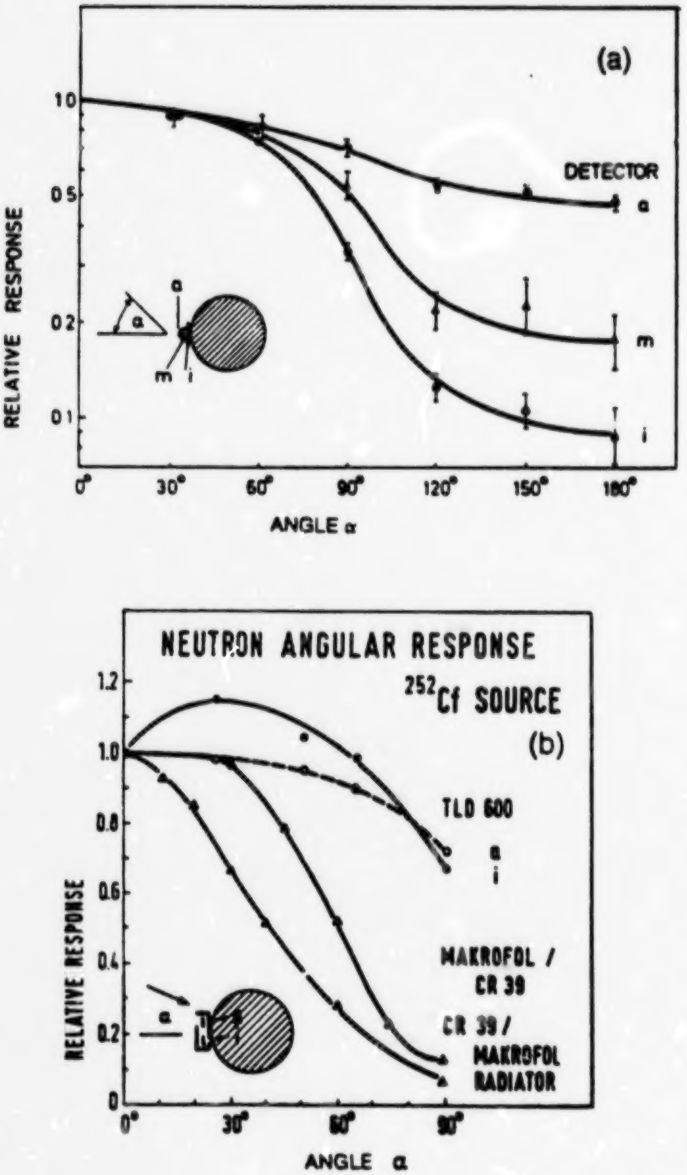
ENERGY DEPENDENCE OF THE ALBEDO NEUTRON DOSIMETER



SOURCE: Piesch and Burgkhardt, 1985

Figure 4.3-2

ANGULAR RESPONSE OF THE ALBEDO NEUTRON DOSIMETER



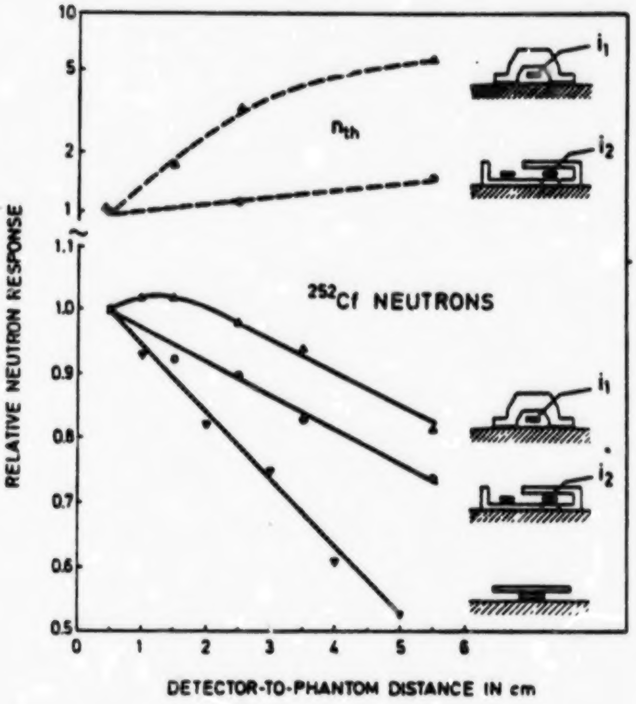
NOTE:

1. Angular response of (a) the Karlsruhe albedo dosimeter, (b) the universal albedo dosimeter with TLD-600 in the Alnor system, and Makrofol and CR-39 recoil track etching detectors electro-chemically etched.

SOURCE: Piesch and Burgkhardt, 1985

Figure 4.3-3

RELATIVE NEUTRON RESPONSE OF ALBEDO NEUTRON DOSIMETERS WITH DISTANCE



NOTE:

1. Albedo detector response plotted against distance of the detector from the body for the Karlsruhe albedo neutron dosimeter, the universal albedo neutron dosimeter, and a TLD-600 detector behind cadmium.

SOURCE: Piesch and Burgkhardt, 1985

4.4 Nuclear Track Emulsions

Hofert and Piesch (1985) reviewed the recent progress that had been made in neutron dosimetry using nuclear film type A (NTA) produced by Kodak. It is the most commonly used personal neutron monitoring film.

The theory of operation is based upon the following. Undeveloped NTA film is exposed in the neutron field and the most significant nuclear reaction that occurs is the collision between neutrons and protons in the emulsion itself, in the plastic base, in the film packaging and in the film holder. The recoil protons that pass through the emulsion "expose" it along the track length. When the film is developed, the proton track is visible for counting under a microscope.

The film can be used for neutron dosimetry only in the neutron energy range 0.5 to 10 MeV. Neutrons below 0.5 MeV impart too little energy to the proton for the latter to make a sufficiently long and visible track. At the other end of the scale, neutrons with more than 10 MeV of energy transfer too much energy to the proton during the collision and the track in the emulsion is too weak to count. Within the 0.5 to 10 MeV range, the sensitivity of the film varies with neutron energy. The dose equivalent energy response is shown in Figure 4.4-1 and represented schematically in Figure 5.1-1.

As with all photographic emulsions, NTA film is particularly sensitive to high heat and humidity between the time of the irradiation of the film and its development. The effect of high humidity is particularly severe. At room temperature and 100% relative humidity (RH), all tracks disappear within 3 days of formation. Fading at other temperatures and humidities is shown in Figure 4.4-2. Fading can be reduced by storing and handling the film at controlled temperatures and humidities and by developing the film as soon after exposure as possible.

The neutron dose range is somewhat dependent upon the neutron energy. For neutrons of 2 to 4 MeV, the lower and upper dose limits are 0.4 and 200 mSv, respectively. For higher energy neutrons near high energy proton accelerators, the upper dose limit falls to about 50 mSv.

Interference from low energy photons is more severe than from higher energy photons. A soft x-ray dose of 0.5 mSv will interfere with the neutron count, while it takes 20 mSv of gamma radiation to interfere with counting. Lead shields placed over the film to reduce photon sensitivity also reduce the neutron sensitivity by preventing recoil protons from the case from reaching the film.

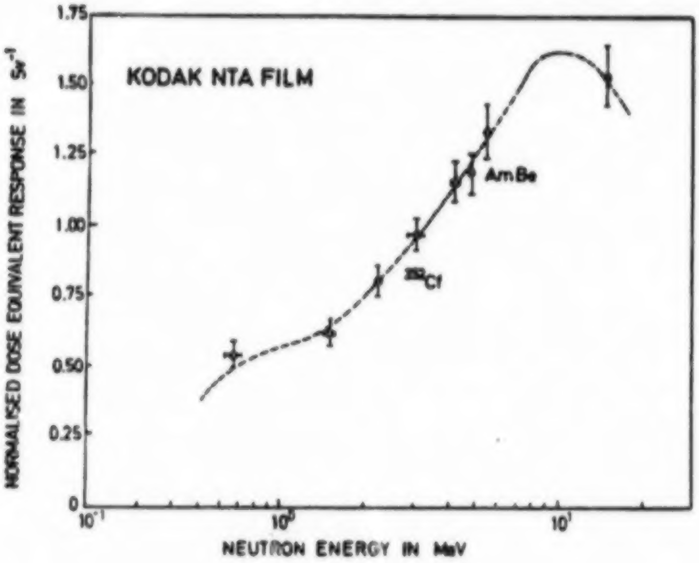
The response of the NTA film to thermal neutrons has been found to be 45% higher than the response to ^{241}Am-Be neutrons. Thermal neutrons can be discriminated against by placing a cadmium filter around the film.

Background tracks are few in number and tend to fade with time. Therefore, they do not limit accuracy or sensitivity.

The overall uncertainty in assigning a dose equivalent based upon an NTA film reading is influenced by many factors such as the uncertainty in the absolute number of background tracks and its variation within the batch and the emulsion as a detector shows some variation in sensitivity even for films of the same production batch. Experiments showed that the standard deviation due to the influence of batch uniformity, the development of the film and the individual track counting statistics were of the order of 10% (Hofert, 1985). Based upon experiments at a dose equivalent of 1 mSv, the standard deviation of error is estimated to be 20%, if the fading can be controlled to 10% over the normal wearing period.

Figure 4.4-1

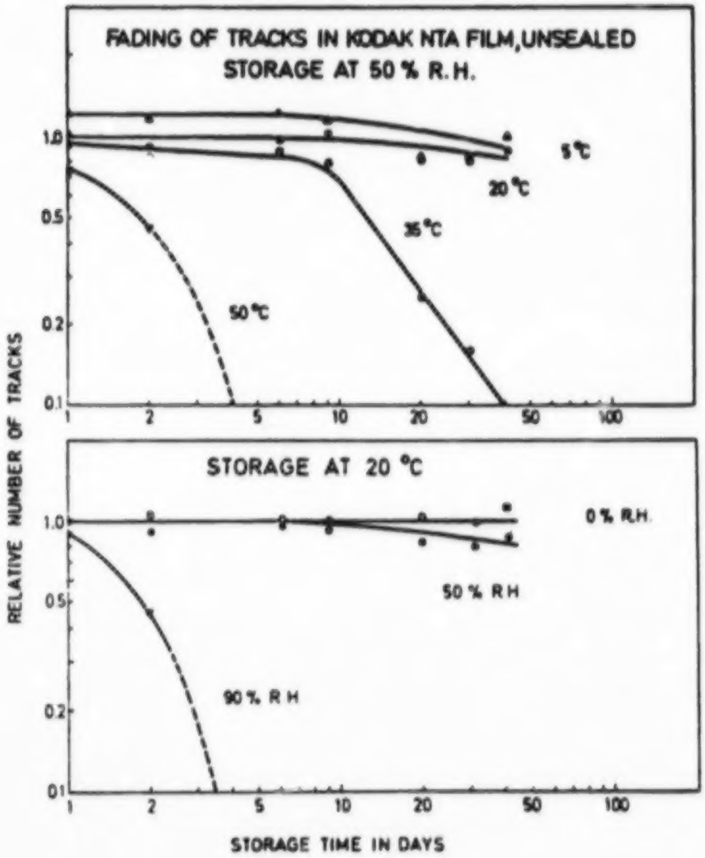
DOSE EQUIVALENT RESPONSE OF NTA EMULSION AS A FUNCTION OF NEUTRON ENERGY



SOURCE: Höfert and Piesch, 1985

Figure 4.4-2

FADING OF NUCLEAR TRACKS AS A FUNCTION OF STORAGE TIME AFTER IRRADIATION



NOTE:

1. Parameter in the upper figure is temperature; that in the lower one is relative humidity (RH). Films were developed in a strongly reducing developer at 29° C.

SOURCE: Hofert and Piesch, 1985

4.5 Resonance Ion Spectroscopy

Resonance ion spectroscopy, as a method of measuring the dose equivalent in a neutron radiation field, is at the very preliminary research stage. Hurst (1981) proposed a method whereby a suitable element, such as krypton, is impregnated into a matrix which is fixed to a polyethylene radiator. Under neutron bombardment, charged particles are knocked off the radiator, strike the matrix and free some of the trapped krypton. The krypton is collected and measured using resonance ion spectroscopy which is sensitive enough to detect a single atom. No recent published information was found concerning this new approach.

4.6 Intercomparison Studies

Swaja and Sims (1988) reported the summary of six intercomparison experiments on neutron dosimeters conducted at Oak Ridge National Laboratory (ORNL) over the years 1981 to 1986. During these studies, personal dosimeters were mailed to ORNL mounted on phantoms, exposed to low-level (less than 15 mSv) dose equivalents in mixed radiation fields mainly produced using the Health Physics Research Reactor (HPRR) at ORNL, and then returned to the participants for evaluation.

Although many different dosimeter designs were submitted for testing, the basic detection mechanisms can be classified into six categories: direct-interaction thermoluminescent (TLD), TLD-albedo, film, recoil track, fission track and combination albedo plus recoil track.

To determine low dose equivalent sensitivity, irradiations were conducted using a variety of HPRR spectra at low dose equivalents of about 0.5 mSv. Four different spectra were generated using filters of steel, concrete and Lucite (Figure 4.6-1) with average neutron energy ranging from 0.78 to 1.306 MeV and neutron/gamma dose equivalent ratios ranging from 11.8 to 62.6. For each dosimeter type, a figure of merit was calculated to be x/y where x is the number of measurements for which a dose estimate was reported to be greater than zero or the minimum detection limit and y equals the total number of measurements. In general, the figure of merit shows that participants who used TLD-albedo systems had fewer problems obtaining measurable indication of neutron exposure (~96%) at about 0.5 mSv than did those who used track based systems (52% to 75% for recoil and fission track systems, respectively). Nonzero dose estimates were determined with a figure of merit of only about 70% for NTA film.

The accuracy of different basic neutron dosimeter systems (Figure 4.6-1) shows normalized results for reference dose equivalents greater than 1.5 mSv as a function of incident HPRR spectra. With regard to overall performance, direct interaction TLDs, albedos and recoil track dosimeters provide average results within about 10% of reference values for the hardest (unshielded) spectrum. However, at softer spectra, TLD-based systems tend to overestimate reference dose equivalents with direct-interaction systems overresponding more than albedos with increasing spectrum softness. Detectors with threshold detection energies such as recoil track on film, underestimate reference dose equivalents for moderated spectra by more than 20%. The average reported results for all the basic dosimeter types included in Figure 4.6-1 are within 40% of reference values with the exception of the direct-interaction TLD for the softest neutron spectrum.

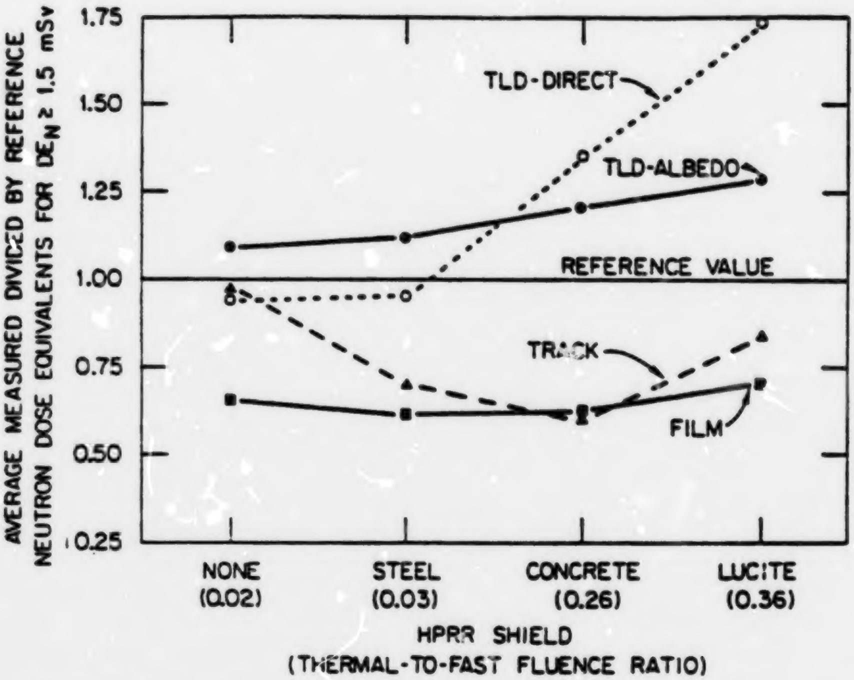
With regard to measurement precision (reproducibility for each measurement for each participant), albedo and direct-interaction TLDs exhibited the best precision with average standard deviations of about 9% of the means. About 75% of all organizations using these dosimeter types had individual standard deviations of 10% or less. Fission track systems produced the poorest precision with average standard deviations of about 18% of the means. Film, recoil track and combination systems showed precisions close to but slightly poorer than results obtained for TLDs.

As part of one of the intercomparisons, several irradiations were conducted using accelerators to obtain neutrons with energies of 5.3 and 15 MeV. At dose equivalent levels about 0.5 mSv all basic neutron dosimeter types exhibited significant variations in results and generally poor accuracy relative to reference values. At levels of 10 mSv, average dose equivalents measured by film neutron dosimeters were within 20% of reference values for both neutron energies (combination albedo-track systems, within 20%; albedo and direct-interaction TLDs, within 60%; CR-39 systems, within 25% and 75% at 5.3 and 15 MeV, respectively). It should be noted that since these tests were conducted in 1982, the dose equivalent energy response of CR-39 up to 15 MeV has been significantly improved.

In summary, the TLD-albedo system had the highest accuracy, precision and low dose sensitivity to neutrons with average energy equal to fission spectra and lower. When the TLD-albedo system was combined with track etch, the combination provided the best accuracy and precision at higher neutron energies (5.3 and 15 MeV).

Figure 4.6-1

ACCURACY OF NEUTRON DOSIMETERS



Neutron dosimeter accuracy as a function of incident spectrum for dose equivalents greater than 1.5 mSv.

Health Physics Research Reactor (HPRR) radiation field characteristics at the dosimeter exposure locations.[a]

HPRR shield	Average neutron energy, MeV	Thermal-to-fast fluence ratio[b]	Neutron-to-gamma dose equivalent ratio
Unshielded	1.306	0.020	62.6
13-cm steel	0.780	0.030	86.6
20-cm concrete	0.885	0.257	22.0
12-cm Lucite	0.951	0.357	11.8

[a] Data at 3 m from the HPRR with the reactor operated over the storage pit at 1.4 m above the floor.

[b] Thermal neutron ($E_n \leq 0.5$ eV) fluence divided by the fast neutron ($E_n \geq 1$ MeV) fluence.

SOURCE: Swaja, 1988

5.0 ASSESSMENT OF BEST SUITED NEUTRON MONITORS AND DOSIMETERS

5.1 Accelerators

5.1.1 Medical Accelerators

The results of measurements of neutron fluence in occupationally accessible areas outside the shielding of medical accelerators showed that there are significant neutron groups in the energy interval 0.1 to 7 MeV and at thermal and intermediate energies. However, mitigating conditions reduce the significance of the two lower energy neutron groups. The dose equivalent conversion factor is more than an order of magnitude higher for the 0.1 to 7 MeV neutrons than it is for the two lower energy groups. Consequently, the dosimetric importance of the thermal and intermediate energy groups is proportionately lower. Secondly, the total neutron dose equivalent rate at the representative facilities reviewed here is maintained well below the appropriate regulatory limits. Under these conditions it is unlikely that personnel in the vicinity of the medical accelerators will receive doses of any significance from the thermal and intermediate neutron groups.

Some neutron monitoring has been carried out at medical facilities in the past. However, none of the testing included the use of albedo dosimeters, the only personal neutron dosimetry system sensitive to thermal and intermediate energy neutrons (Figure 5.1-1). Therefore, it would be prudent to monitor for the thermal and intermediate neutrons until the results of a study proposed in Chapter 6 become available.

Based upon the neutron energy response functions shown in Figure 5.1-1, there is no single personal dosimetry system which can cover all neutron energies encountered near medical accelerators.

Therefore, two or more detector systems will have to be utilized.

The albedo system is an obvious choice because it is the only system which has adequate sensitivity in the thermal and intermediate neutron energy regions. The Superheated Drop Detector in its present form is not suitable for application in a pulsed beam environment. The pulsed neutron source would cause pulse pile-up problems in the sonic detector and the integrated gas detector is not commercially available. The low energy end of the NTA film response function does not overlap with the high energy end of the albedo system so the former is also not suitable. Either the bubble detector or the CR-39 personal dosimeter should be acceptable. The former has the advantages of direct reading capability and lower limit of sensitivity.

Researchers at Princess Margaret Hospital have demonstrated that an Andersson-Braun rem meter can be operated in a pulsed beam medical accelerator environment and will not respond to either background gamma or RF interference. As a survey meter, the Leake type instrument would be the most suitable as it is has a better dose equivalent energy response than the Andersson-Braun and is more portable. Although the Bonner spheres provide much broader neutron energy coverage, they are totally impractical as a survey instrument. Provided the count rate is maintained at a sufficiently low level (section 3.2-2), the Leake rem meter can be used in a pulsed beam radiation environment.

5.1.2 Research Accelerators

Researchers at TRIUMF have made detailed measurements of neutron spectra around the 500 MeV cyclotron using Bonner spheres and the $^{12}C(n,2n)^{11}C$ reaction. They have demonstrated that in some locations the average neutron energy is very high and in others the neutron spectrum consists of thermal neutrons only. Also, in some areas neutrons with energies characteristic of iron leakage are predominant. Consequently, it is essential that the personal neutron dosimetry system be capable of measuring the dose equivalent from neutrons of all energies from thermal up to 500 MeV. As above, the albedo dosimetry system is appropriate for measuring the lower energy end of the spectrum and CR-39 or bubble detectors are the best available for the higher energies. Unfortunately, there is no personal dosimetry system available that can measure neutrons above 20 MeV.

Equipment needed to adequately carry out a neutron survey of the TRIUMF facility includes the Bonner spheres for the energy interval from thermal to about 20 MeV and the carbon scintillator for higher energy neutrons.

5.2 Nuclear Reactors

Neutron spectra measured in reactor halls of pressurized water reactors, and in the boiler and pump rooms of heavy water moderated reactors illustrate a wide diversity of neutron energy groups. Some areas have only slightly moderated fission spectra leaking through shielding, while other areas are exposed to predominantly thermal neutrons. Experimental reactors, such as at McMaster University can purposefully extract neutrons at 24 keV from beam ports through the iron "window". Consequently, the requirements for personal dosimetry include the use of the albedo dosimetry system to cover the thermal and intermediate neutron energies. Neutrons from the top of the albedo response curve to about 20 MeV should be monitored with the CR-39, bubble detector or Superheated Drop Detector systems.

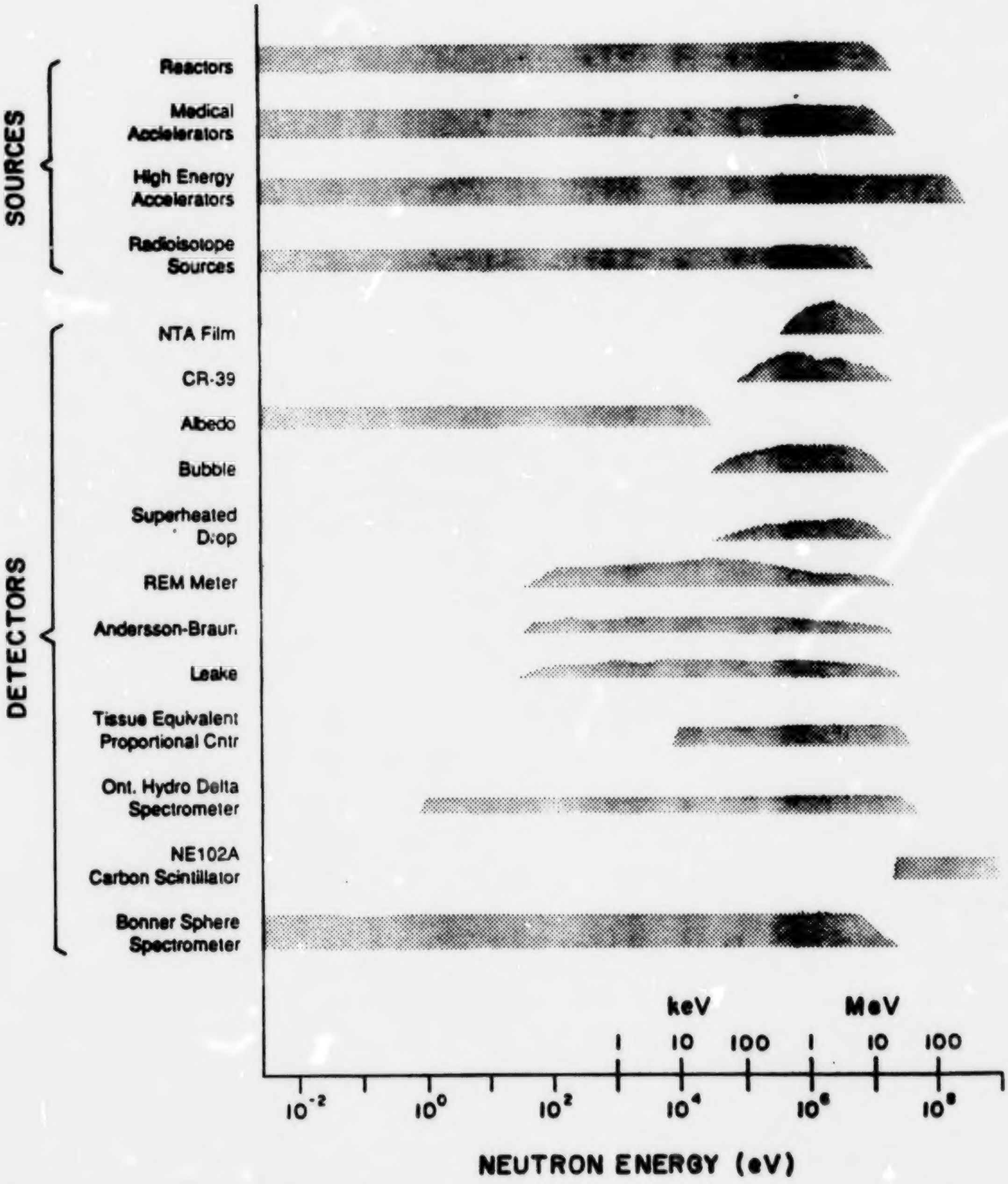
Equipment required to adequately carry out a neutron survey and spectrum analysis near experimental and power reactors includes the Bonner sphere system. Slowpoke reactors may be an exception to this guideline should results of the neutron survey recommended in the next chapter indicate that only thermal neutrons are dosimetrically significant.

5.3 Radioisotopic Neutron Sources

The primary neutron energy generated by radioisotopic neutron sources can have a wide range of values from about 12 MeV to energies as low as 100 keV from ^{238}Pu-Li and 25 keV from ^{124}Sb-Be. For those radioisotopic sources which emit significant numbers of neutrons at energies below 100 keV, personal dosimetry by means of NTA film, CR-39 track etch and even the bubble detector and superheated drop is suspect. The most appropriate personal dosimeter is an albedo system.

FIGURE 5.1-1

NEUTRON SOURCES AND DETECTOR RESPONSES



	Hatched Area	Open Area
Neutrons in the energy range produced by source or scattered from higher energies	Yes	No
Detector sensitivity to neutrons in the energy range	Yes	No or Unknown

6.0 RECOMMENDATIONS

During the review of the literature and the facility information that was provided for this study, various inadequacies in existing portable survey instruments and personal dosimetry systems became apparent. In the following discussion where possible, the deficiencies are identified and a course of action is recommended. Items of high priority or potential for success are listed first. However, the remaining recommendations are grouped by subject rather than by priority.

At the present time, there are no personal dosimetry systems which have the neutron energy response required to monitor neutrons with energies in excess of 20 MeV. Both TRIUMF and the [illegible]atchewan Accelerators produce neutrons with energies above 20 [illegible] ne of the top priorities should be the development of a system capable of measuring doses from neutrons with energies greater than 20 MeV.

Tissue equivalent proportional counters have potential for the direct measurement of neutron dose equivalent in real time. Miniaturized versions have been assembled as personal neutron dosimeters and have the capability to measure to low neutron energies. A test program to evaluate its potential for direct measurement of dose equivalent is recommended.

Hankins and Westermark (1987) have made preliminary studies of the use of track size in electrochemically etched CR-39 foils to infer neutron spectra. They have met with some success and established that the track diameter is related to the neutron energy. Further development in this area should be addressed. Physical parameters influencing track size and the development of the hardware and computer software to manipulate the track image data for analysis would be worthwhile undertakings.

Piesch has observed that the calibration factor for albedo dosimeters can vary by a factor of 10 depending upon the mean value of the neutron energy. He recommends that the use of a two-component albedo system that can separate albedo neutrons and incident field thermal neutrons be further investigated. Preliminary studies suggest that the variation of the calibration factor can be reduced to +/- 30%.

High energy cyclotron accelerators pulse the beam at frequencies of the order of 1 kHz. However, there is usually a higher frequency bunching of the beam at frequencies in the range of tens of MHz. The influence, if any, that the high frequency bunching has on neutron monitoring techniques should be investigated.

At some beam lines at TRIUMF, meson fluxes are generated. Their influence on neutron detection equipment should be investigated to determine whether the mesons represent an interference to neutron detection.

Because the body of the dosimeter wearer distorts the spectral shape and magnitude of the neutron fluence, it has been proposed that two or three sets of different dosimeters should be positioned on the wearer's body. This concept would seem worthwhile investigating. Several aspects could be examined such as, which dosimeter types are best suited for multipoint application, which locations on the wearer are the most appropriate and do the appropriate dosimeter types and locations change with neutron source type.

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